2. Proposed Action and Alternatives

This chapter states the proposed action and describes the management alternatives analyzed in this Environmental Impact Statement (EIS) for implementation of the proposed action. Environmental and policy impacts from the management alternatives are presented in Chapter 4.

2.1 Overview of the Proposed Action and Alternatives

The U.S. Department of Energy (DOE) and Department of State are jointly proposing to adopt a policy to manage spent nuclear fuel from foreign research reactors. Only spent nuclear fuel containing uranium enriched in the United States would be covered by the proposed action. The purpose of the proposed action is to promote U.S. nuclear weapons nonproliferation policy objectives, specifically by seeking to reduce, and eventually eliminate, highly-enriched uranium (HEU) from civilian commerce.

To implement the proposed action, DOE and the Department of State have considered three foreign research reactor spent nuclear fuel management alternatives. They are:

1. To accept and manage foreign research reactor spent nuclear fuel in the United States (Management Alternative 1);
2. To facilitate the management of foreign research reactor spent nuclear fuel at one or more foreign locations (Management Alternative 2); and
3. A combination of elements from Management Alternatives 1 and 2 (Management Alternative 3, Hybrid Alternative).

The management alternatives of the proposed action are portrayed in Figure 2-1 and are discussed in more detail in Sections 2.2, 2.3, and 2.4.

A No Action Alternative on the part of DOE and the Department of State to address the status of the foreign research reactor spent nuclear fuel has also been considered in this EIS. The No Action Alternative is discussed in Section 2.5.

DOE and the Department of State have identified a preferred alternative for the proposed action. The preferred alternative is described in Section 2.9.

The foundation for the analysis presented in Chapter 4 of this EIS is the evaluation of the components that comprise the basic implementation of Management Alternative 1. The basic implementation concept is an attempt by DOE and the Department of State to avoid unnecessary repetition by selecting a reasonable option for each component and examining them in detail under Management Alternative 1. Since the No Action Alternative would not have any direct environmental impacts in the United States, it requires only policy analysis in this EIS. Management Alternatives 1, 2, and 3, however, would all have environmental impacts in the United States, and the components of the basic implementation provide the parameters with which to analyze their potential environmental impacts in this EIS.

The detail of analysis provided for the basic implementation components is based on the fact that some variation of these components is utilized in each implementation alternative under Management Alternative 1, as well as in Management Alternative 3. In this way, analysis of the implementation

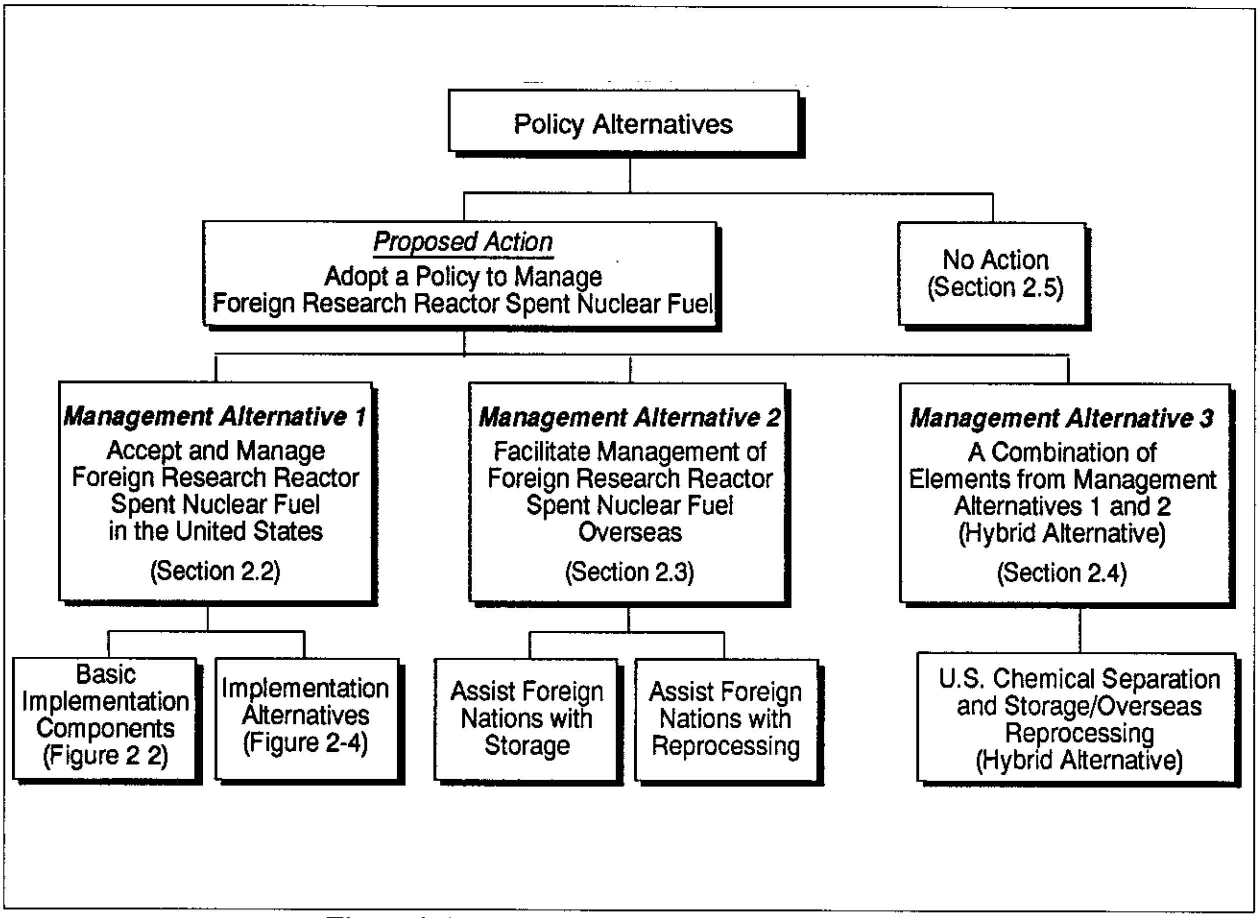

Figure 2-1 Policy and Management Alternatives

alternatives, as well as Management Alternative 3, can be tiered from the analysis of the basic implementation. In and of itself, the basic implementation of Management Alternative 1 is a viable implementation alternative for consideration under Management Alternative 1, along with the other implementation alternatives discussed below. However, the level of detail contained in the analysis of the basic implementation does not indicate any preference for this alternative. Rather, it merely eliminates the need to duplicate information later in the analysis.

The components of the basic implementation of Management Alternative 1 would consist of the following:

1. A policy duration of 10 years.

2. A financing arrangement by which the United States would bear the full cost for transporting and managing the foreign research reactor spent nuclear fuel received from developing countries, but would charge developed countries a competitive fee.

3. The receipt of a fixed amount of foreign research reactor spent nuclear fuel containing uranium enriched in the United States. This fixed amount is up to approximately 22,700 foreign research reactor spent nuclear fuel elements and is based on estimated inventories of foreign research reactor spent nuclear fuel currently stored or to be generated in the 10-year policy period.

4. Taking title to the foreign research reactor spent nuclear fuel at the U.S. territorial waters limit (19 km or 12 mi), or continental U.S. borders for shipments from Canada.

5. Marine transport of the foreign research reactor spent nuclear fuel by chartered and/or regularly scheduled commercial ships.

6. Ports of entry that qualify on the basis of criteria discussed in this EIS.

7. Ground transport from ports of entry to storage sites, and between sites (by truck, rail, or barge, or a combination of these modes.)

8. Potential storage sites identified in the Programmatic SNF&INEL Final EIS (DOE, 1995c) for foreign research reactor spent nuclear fuel, namely the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site.

9. Use of dry storage technology for construction of new storage facilities.

The basic implementation components are depicted in Figure 2-2 and described in Section 2.2.1. Environmental impacts and policy considerations of the basic implementation components of Management Alternative 1 are presented in Section 4.2.

Utilizing the components provided above, DOE has evaluated seven implementation alternatives for Management Alternative 1 in addition to the basic implementation. Each implementation alternative is comprised of the same components as the basic implementation; however, for the purpose of analysis, one of the components has been varied. The seven implementation alternatives are given below.

1. Acceptance of amounts of foreign research reactor spent nuclear fuel different from the amounts identified in the basic implementation.

2. Acceptance of foreign research reactor spent nuclear fuel for a period of time different from the time period identified in the basic implementation.

3. Financial arrangements different from those identified in the basic implementation.

4. Taking title to foreign research reactor spent nuclear fuel at locations different from those identified in the basic implementation.

5. Use of wet storage technology for construction of new storage facilities instead of dry storage technology as identified in the basic implementation.

6. Use of near term conventional chemical separation in the United States to reduce the duration of, and amount of, spent nuclear fuel storage required.

7. Use of new developmental treatment and/or packaging technologies in addition to storage as identified in the basic implementation.

Implementation alternatives for Management Alternative 1 are discussed in Section 2.2.2. The environmental impacts and policy considerations of the implementation alternatives are discussed in Section 4.3.

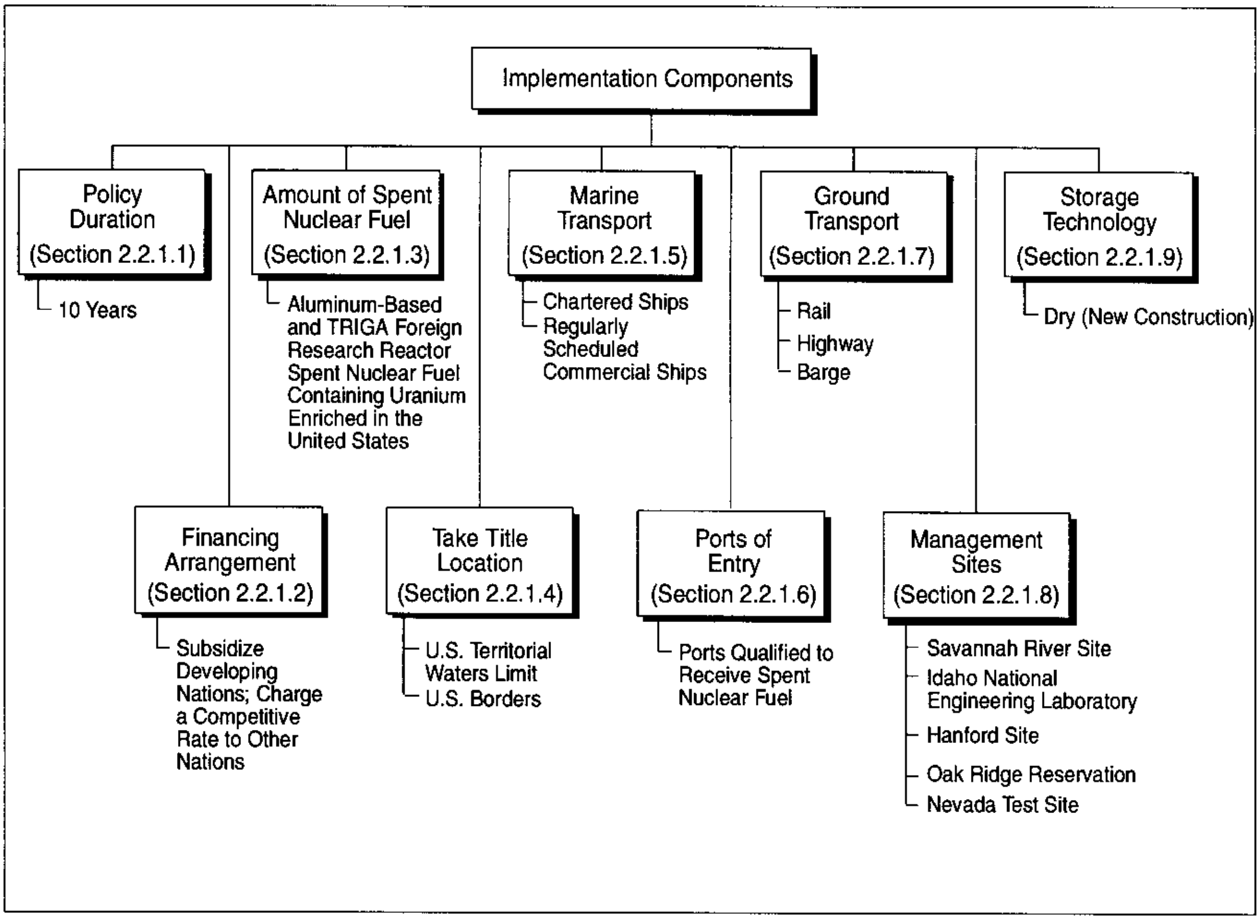

Figure 2-2 Basic Implementation Components

Qualifying Fuel Types and Policy Stipulations

This policy applies solely to aluminum-based and TRIGA[1] research reactor fuels and target materials containing HEU and low enriched uranium (LEU) of U.S. origin. Aluminum-based fuel is clad in aluminum and has an active fuel region that consists of an alloy of uranium and aluminum or a dispersion of uranium-aluminide, uranium-oxide[2] or uranium-silicide in aluminum. TRIGA fuel consists of an alloy of uranium and zirconium and is clad in either aluminum or stainless steel. Fuels containing significant quantities of Uranium-233 (^{233}U) are excluded. Target materials are the residual materials from isotope production targets in research reactors.

The policy would include the following stipulations:

- Spent nuclear fuel (either and/or both HEU and LEU) would be accepted from research reactors operating on LEU fuel or in the process of converting to LEU fuel when the policy becomes effective.

1 TRIGA stands for Training, Research, Isotope, General Atomic reactors.

2 This uranium-oxide composition refers to aluminum-clad fuel plates or tubes containing dispersions of U_3O_8 in aluminum. It does not include fuels containing UO_2 pellets clad in aluminum, zircaloy, stainless steel, or other materials, or uranium-silicide in aluminum.

- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors which operate on HEU fuel when the policy becomes effective and which agree to convert to LEU fuel. Spent nuclear fuel would not be accepted from research reactors that could convert to LEU fuel but refuse to do so.
- Spent nuclear fuel (HEU) would be accepted from research reactors having lifetime cores, from research reactors planning to shut down by a specific date while the policy is in effect, and from research reactors for which a suitable LEU fuel is not available.
- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors that are already shut down.
- Unirradiated fuel (HEU and/or LEU) from eligible research reactors would be accepted as spent nuclear fuel.
- For research reactors with both HEU and LEU spent nuclear fuel available for shipment, LEU spent nuclear fuel would not be accepted until the HEU spent nuclear fuel is exhausted, unless there are extenuating circumstances (e.g., deterioration of one or more LEU elements sufficient to cause a safety problem).
- Spent nuclear fuel (HEU and/or LEU) would not be accepted from new research reactors starting operation after the date of implementation of the policy.

Ultimate Disposition

Ultimate disposition of DOE's spent nuclear fuel, including foreign research reactor spent nuclear fuel, is a high priority. For planning purposes, DOE has determined that its spent nuclear fuel that is not otherwise managed (e.g., chemically separated, with the high-level waste being converted into a vitrified glass for repository disposal) is authorized for disposal in a geologic repository. The Nuclear Waste Policy Act of 1982 (as amended) authorizes disposal of the foreign research reactor spent nuclear fuel in a geologic repository (if DOE takes title to such spent nuclear fuel). However, since the repository characterization program is in its early stages, the waste acceptance criteria for deposit of DOE's spent nuclear fuel in a repository have not been developed. Thus, a determination cannot be made at this time as to the requirements that must be met to allow placement of the foreign research reactor spent nuclear fuel in the repository. As a result, the EIS analysis for the time period beyond 40 years is qualitative rather than quantitative. The qualitative assessment includes consideration of disposal of intact foreign research reactor spent nuclear fuel, disposal of vitrified high-level waste resulting from chemical separation, as well as utilization of various potential new technologies to process the spent nuclear fuel into a more stable form prior to its ultimate disposition. In the event that the geologic repository schedule is delayed beyond the 40-year program period, DOE would continue to manage the foreign research reactor spent nuclear fuel or any resultant stable waste forms in existing facilities at the DOE management site(s) until a geologic repository becomes available. Decisions regarding the actual disposition of DOE's spent nuclear fuel will follow appropriate review under the National Environmental Policy Act (NEPA).

2.2 Management Alternative 1 - Manage Foreign Research Reactor Spent Nuclear Fuel in the United States

2.2.1 Basic Implementation Components

2.2.1.1 Policy Duration

The policy duration of the basic implementation of Management Alternative 1 would be 10 years, beginning on the date when the management policy becomes effective. Spent nuclear fuel containing HEU and LEU of U.S. origin that is currently being stored or is to be generated during the 10-year period of the policy would be accepted.

Actual shipments of spent nuclear fuel to the United States could be made for a period of 13 years starting from the effective date of the policy implementation, as long as the spent nuclear fuel was generated within the 10-year policy period. The 3 additional years would allow for a cooling time for fuel discharged from a reactor late in the policy period, logistics in arranging for shipment of this fuel, as well as other unplanned for potential delays.

2.2.1.2 Financing Arrangements

The United States would bear the full cost of transporting and managing the foreign research reactor spent nuclear fuel received from developing countries. Developing countries are defined by the World Bank as those countries having other than high-income economies (World Bank, 1994). For developed countries, however, the United States would charge a competitive fee for the handling, storage, conditioning (as needed), and any disposal activities conducted by the United States. Tables 2-1 and 2-2, which provide estimates of the number of elements that may be accepted, identify those countries defined as developing countries.

2.2.1.3 Amount of Foreign Research Reactor Spent Nuclear Fuel

The analysis in this EIS is based primarily on the number of individual elements of foreign research reactor spent nuclear fuel that could be accepted. When appropriate, the analysis also uses two other measures to express the amounts in understandable terms:

- Mass of Heavy Metal. This is the mass of all the heavy metal atoms in the spent nuclear fuel (mostly uranium), excluding the mass of other materials such as alloys, cladding, and structural materials. The international standard unit of measure for this quantity is metric tons of heavy metal (MTHM).
- Volume. The volume of the spent nuclear fuel is important because it determines the number of shipments and the storage space required. The volume is expressed in cubic meters.

The amount of foreign research reactor spent nuclear fuel that would be accepted under the basic implementation is up to approximately 19.2 MTHM, with a volume of approximately 110 m^3 (4,100 ft^3), representing approximately 22,700 individual spent nuclear fuel elements. The number of elements cited for acceptance under the policy includes those elements at issue in the Environmental Assessment of Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel (DOE, 1994m).

Table 2-1 Estimated Number of Aluminum-Based Spent Nuclear Fuel Elements Generated by Foreign Research Reactor Operators by January 2006

Country	Estimated Number of Spent Nuclear Fuel Elements	Initial Mass of Uranium (MTHM)[a]	Estimated Number of Shipments
Argentina[b]	283	0.071	9
Australia	975	0.427	9
Austria	157	0.191	5
Belgium	1,766	0.730	59
Brazil[b]	155	0.099	5
Canada	2,831	4.478	116
Chile[b]	58	0.012	2
Colombia[b]	16	0.002	1
Denmark	660	0.529	22
France	1,962	3.442	149
Germany	1,504	0.909	49
Greece[b]	239	0.113	8
Indonesia[b]	198	0.236	6
Iran[b]	29	0.006	1
Israel	192	0.111	6
Italy	150	0.043	5
Jamaica[b]	2	0.001	1
Japan	2,981	3.134	99
Korea (South)[b]	168	0.321	7
Netherlands	1,488	1.404	49
Pakistan[b]	82	0.016	3
Peru[b]	29	0.039	1
Philippines[b]	50	0.024	2
Portugal[b]	88	0.054	3
South Africa[b]	50	0.010	2
Spain (from Scotland)[c]	40	0.016	1
Sweden	1,113	1.374	37
Switzerland	159	0.128	5
Taiwan	127	0.066	4
Thailand[b]	31	0.005	1
Turkey[b]	69	0.089	2
United Kingdom	12	0.004	1
Uruguay[b]	19	0.018	1
Venezuela[b]	120	0.082	4
Total	17,803	**18.184**	**675**

[a] *To derive uranium mass in kilograms, multiply the amounts by 1,000.*

[b] *Countries with other than high-income economies (World Bank, 1994).*

[c] *40 Spent nuclear fuel elements of Spain's JEN-1 Reactor core are stored in Dounreay, Scotland.*

Implementation of Management Alternative 1 would involve less than 1 percent of the total mass of heavy metal that DOE currently manages as spent nuclear fuel (DOE, 1994c), and approximately 10 percent of the volume.

Table 2-2 Estimated Number of TRIGA[a] Reactor Spent Nuclear Fuel Elements Generated by Foreign Research Reactor Operators by January 2006

Country	*Estimated Number of Spent Nuclear Fuel Elements*	*Initial Mass of Uranium (MTHM)[b]*	*Estimated Number of Shipments*
Austria	106	0.020	3
Bangladesh[c]	100	0.049	3
Brazil[c]	75	0.014	3
Finland	171	0.033	6
Germany	358	0.068	12
Indonesia[c]	245	0.047	8
Italy	386	0.072	13
Japan	326	0.062	11
Korea (South)[c]	336	0.064	11
Malaysia[c]	94	0.047	3
Mexico[c]	186	0.035	6
Philippines[c]	128	0.079	4
Romania[c]	1,451	0.189	48
Slovenia[c]	393	0.075	13
Taiwan	144	0.086	5
Thailand[c]	136	0.035	4
Turkey[c]	79	0.015	2
United Kingdom	90	0.017	3
Zaire[c]	136	0.026	4
Total	**4,940**	**1.033**	**162**

[a] *TRIGA is an acronym for Training, Research, Isotope, General Atomic reactors.*

[b] *To derive uranium mass in kilograms, multiply the amounts by 1,000.*

[c] *Countries with other than high-income economies, developing countries (World Bank, 1994)*

The Notice of Intent [59 Fed. Reg. 54336 (1993)] for this EIS estimated that 15,000 spent nuclear fuel elements would be accepted under the proposed action. This estimate [representing 12 MTHM, with a volume of approximately 89 m^3 (3,200 ft^3)] was prepared in early 1993, based on a projected 10-year period of generation of spent nuclear fuel at foreign research reactors in 28 foreign countries, plus the spent nuclear fuel available at these foreign research reactors as of 1993.

Since preparation of the 1993 spent nuclear fuel projection, however, cooperative understandings have been reached with several other foreign research reactor operators concerning their participation in the proposed spent nuclear fuel management program. In addition, the period of time over which the management policy would be in effect has been delayed by 3 years (to mid-1996) and thus at least 3 more years' worth of spent nuclear fuel has accumulated. Thus, the amount of spent nuclear fuel from foreign research reactors that would be accepted under the basic implementation is increased to a new total of up to 19.2 MTHM, with a volume of approximately 110 m^3 (4,100 ft^3), representing approximately 22,700 spent nuclear fuel elements of the type considered in the 1993 projection. Of this amount, approximately 4.6 MTHM is HEU, and 14.6 MTHM is LEU foreign research reactor spent nuclear fuel.

Tables 2-1 and 2-2 provide an estimate of the amount of spent nuclear fuel that has been or would be generated in each country by late 2005 (10 years from the effective date of the policy implementation), as estimated by Argonne National Laboratory based on information provided by the foreign research reactor operators (Matos, 1994). A list of the foreign research reactors included in the proposed policy is provided in Appendix B. Table 2-1 shows the inventory of aluminum-based fuel clad in aluminum, while Table 2-2 shows zirconium-based TRIGA fuel clad in either aluminum or stainless steel. These two tables are combined to yield the approximately 22,700 elements (or about 19.2 MTHM) that are estimated to be currently stored or generated by the year 2005. The tables also provide the estimated number of shipments expected from each country. The number of shipments is a key parameter in evaluating the risks associated with the handling and transportation of the foreign research reactor spent nuclear fuel.

It should be noted that the number of elements and number of shipments presented for each country in Tables 2-1 and 2-2 are estimates based on projections of the numbers of elements to be generated over a ten-year period into the future. These estimates are intended to conservatively bound the total number of foreign research reactor spent nuclear fuel elements and shipments associated with the proposed policy. However, the actual distribution of elements and shipments among the listed countries might change, within the limits of the total number of elements and shipments listed, based on actual experience gained during the lifetime of any policy that may be established.

For the purpose of analysis, the foreign research reactor spent nuclear fuel has been categorized by fuel type (aluminum-based or TRIGA) and geography (Eastern or Western) depending on the location of the likely port(s) of entry to the United States. As noted in Section 2.6.4.1, foreign research reactor spent nuclear fuel from Europe, Africa, and the Middle East and parts of Central and South America is likely to enter the United States from the east coast (Eastern) and the rest from the west coast (Western).

The distribution of the foreign research reactor spent nuclear fuel under the basic implementation is as follows:

- By Fuel Type: Aluminum-based — approximately 17,800 elements, 18.2 MTHM, 105 m^3 (3,900 ft^3)

  TRIGA — approximately 4,900 elements, 1.0 MTHM, 5 m^3 (200 ft^3)

- By Geography: Eastern — approximately 16,400 elements, 14.4 MTHM, 80 m^3 (3,000 ft^3)

  Western — approximately 6,300 elements, 4.8 MTHM, 30 m^3 (1,100 ft^3)

The assumptions used in estimating the number of shipments are included in Appendix B (Section B.1.6). Characteristics of the foreign research reactor spent nuclear fuel that would be accepted are provided in Section 2.6.1.

2.2.1.4 Location for Taking Title to Foreign Research Reactor Spent Nuclear Fuel

Under the basic implementation of Management Alternative 1, DOE would take title to the foreign research reactor spent nuclear fuel at the limit of U.S. territorial waters, or continental U.S. borders for shipments from Canada. Where DOE takes title would not have an effect on the environment. Title location of the spent nuclear fuel is relevant to questions that include the source and extent of liability for damage in the event of an accident outside the scope of Price-Anderson Act coverage. The Price-Anderson Act [42 U.S.C. §2210 (1988)] provides a mechanism by which DOE could pay for damages arising out of a nuclear incident that occurs within the United States.

2.2.1.5 Marine Transport

Under the basic implementation of Management Alternative 1, the foreign research reactor spent nuclear fuel would be shipped by chartered and/or regularly scheduled commercial ships from foreign ports to the United States. Chartered shipments would be on purpose-built ships or general purpose commercial cargo ships meeting appropriate International Marine Organization regulations. Regularly scheduled commercial shipments would be on general purpose commercial ships carrying other cargo at the same time.

Marine transport of the foreign research reactor spent nuclear fuel, as well as ground transport between ports and management sites in the United States, would be carried out in approved and certified spent nuclear fuel casks. These casks would be certified for use in the United States by the U.S. Department of Transportation if the cask was designed and fabricated in a foreign country, or by the U.S. Nuclear Regulatory Commission (NRC) if the cask was designed and fabricated in the United States. The design and fabrication of casks in a foreign country is based on the International Atomic Energy Agency standards which are the bases for those promulgated by the NRC. Marine transport activities are discussed in more detail in Section 2.6.3.

All ships entering U.S. territorial waters are required to comply with U.S. Coast Guard safety regulations and are subject to U.S. Coast Guard inspection. In addition, international transportation of hazardous material is governed by the International Movement of Dangerous Goods Code, which is one of a series of safety codes associated with the International Maritime Organization. This code establishes the international rules for shipping hazardous cargos, which includes foreign research reactor spent nuclear fuel. While most nations have agreed to follow the International Maritime Organization codes, including the International Movement of Dangerous Goods Code, compliance by individual shippers would be voluntary.

Unless DOE were to take title to the foreign research reactor spent nuclear fuel overseas (Section 2.2.2.4), the responsibility for shipping the spent nuclear fuel to the United States (if the fuel is to be accepted into the United States) belongs to the foreign research reactor operators. Under these conditions, DOE would ensure that the shipment of the spent nuclear fuel was accomplished on well-equipped, -maintained, and -operated ships through the contract that would be signed between DOE and every participating foreign research reactor operator. DOE would require the use of carriers that commit to following the International Movement of Dangerous Goods Code and all other safety requirements, such as the Safety of Life at Sea, through these contracts. If DOE were to be responsible for shipping, only shipping firms that guaranteed to follow U.S. Coast Guard regulations and international safety codes would be used to ship foreign research reactor spent nuclear fuel.

2.2.1.6 Port(s) of Entry

The basic implementation of Management Alternative 1 would involve receipt of foreign research reactor spent nuclear fuel at any of the 10 ports of entry chosen on the basis of criteria discussed in Section 2.6.3.1. All 10 candidate ports offer standard cargo container unloading services. These potential ports of entry have been identified subsequent to the application of criteria, (including appropriate experience, safe transit, adequate facilities, and population) to the universe of potential U.S. marine ports of entry. These ports are: Charleston, SC (includes Naval Weapons Station [NWS] Charleston and Wando Terminal); Galveston, TX; Hampton Roads, VA (includes Newport News, Norfolk, and Portsmouth terminals); Jacksonville, FL; Military Ocean Terminal Sunny Point (MOTSU), NC; NWS Concord, CA; Portland, OR; Savannah, GA; Tacoma, WA; and Wilmington, NC. The geographic location of each of these ports is displayed in Figure 2-3. This EIS will also assess the potential impacts of foreign

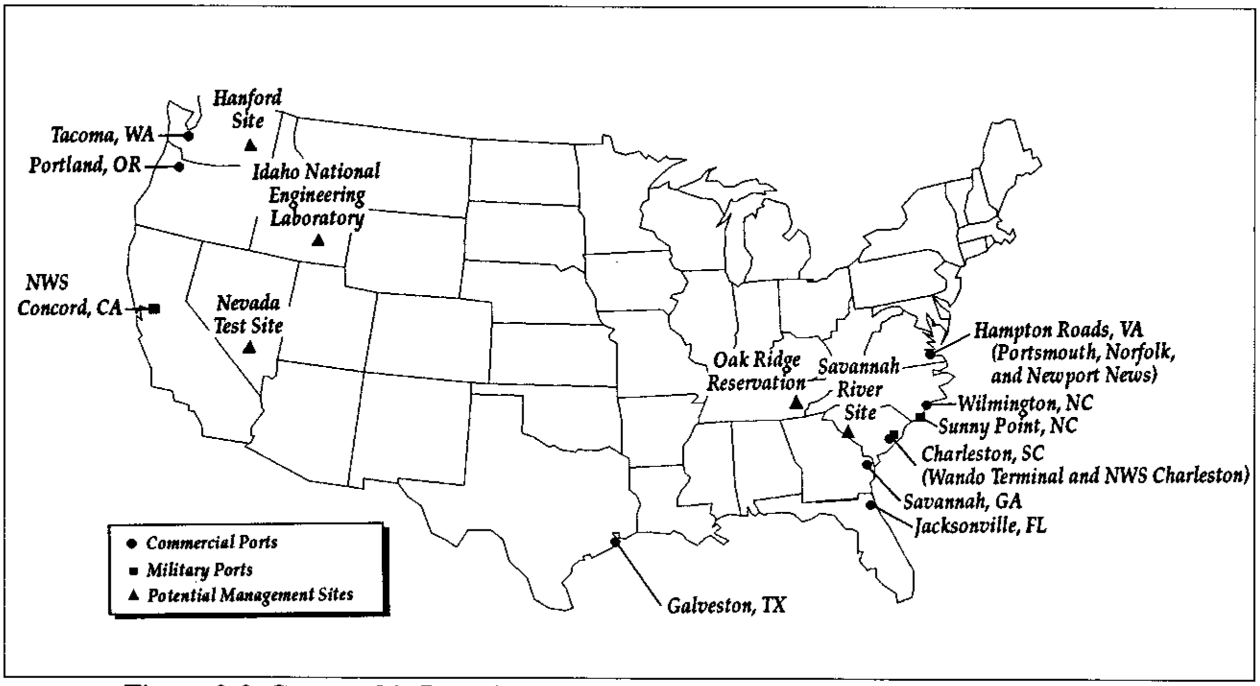

Figure 2-3 Geographic Locations of the Ports of Entry Considered for Receipt of Foreign Research Reactor Spent Nuclear Fuel

research reactor spent nuclear fuel at three high-population-density ports to bound the results of the impact analysis. These high-population-density ports are: Elizabeth, NJ; Long Beach, CA; and Philadelphia, PA. The port identification and evaluation process is discussed in Section 2.6.3 and in Appendix D.

2.2.1.7 Ground Transport

The basic implementation of Management Alternative 1 would involve shipment of foreign research reactor spent nuclear fuel from the ports of entry (both seaports and Canadian border crossings) to potential management sites. It could also involve shipment of foreign research reactor fuel between management sites. As explained in Section 2.6.4.1, the unavailability of certain sites to accept foreign research reactor spent nuclear fuel at the beginning of the management policy period would necessitate temporary receipt and management of foreign research reactor spent nuclear fuel at an available site and subsequent transportation to another site. The ground transport options and route identification process are discussed in Section 2.6.4.

Both rail and highway shipping capabilities are available at all ports of entry and each management site under consideration, with the exception of the Nevada Test Site, which has no rail capability. The shipment of foreign research reactor spent nuclear fuel was analyzed along representative highway and railway routes between all ports and the potential management sites as applicable. Barge transportation is also considered where applicable. The only management sites reasonably accessible by barge are the Savannah River Site and the Hanford Site from the ports of Savannah, GA and Portland, OR, respectively.

2.2.1.8 Foreign Research Reactor Spent Nuclear Fuel Management Sites

Potential sites considered by DOE for the receipt and management of foreign research reactor spent nuclear fuel under this EIS are the same as those considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c). They are the Savannah River Site, the Idaho National Engineering Laboratory, the

Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site. Site-specific activities associated with the basic implementation of Management Alternative 1 and the implementation alternatives are discussed in Section 2.6.5. There are only two sites in the United States that could start receiving these shipments quickly: the Savannah River Site and the Idaho National Engineering Laboratory.

2.2.1.9 Storage Technologies

Under the basic implementation of Management Alternative 1, DOE would receive and manage foreign research reactor spent nuclear fuel for a period starting in approximately mid 1996, and continuing for 40 years until ultimate disposition. During the first few years, storage would take place in existing storage facilities that use both wet and dry storage technologies. For the period beyond those first few years, when construction of new facilities may become necessary, the storage technology identified for the basic implementation of Management Alternative 1 is dry storage. However, construction of new wet storage facilities is considered as an implementation alternative. Storage technologies and storage facilities considered under the basic implementation of Management Alternative 1 and implementation alternatives are discussed in Section 2.6.5.

2.2.2 Implementation Alternatives for Management Alternative 1

Environmental effects of each of the implementation alternatives are evaluated in this EIS. The range of these alternatives is presented in Figure 2-4. Chapter 4, Section 4.3, provides results of the analysis.

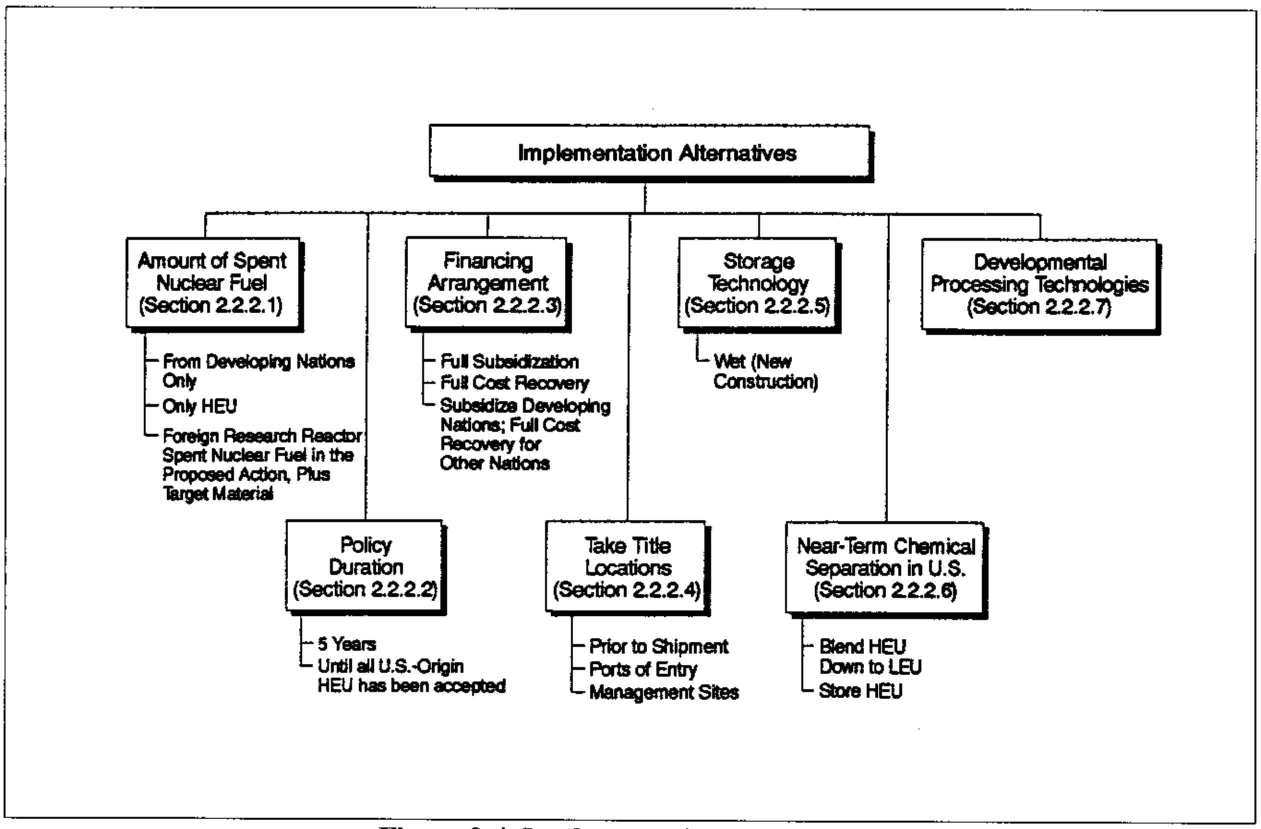

Figure 2-4 Implementation Alternatives

2.2.2.1 Implementation Alternative 1 - Alternative Amounts of Spent Nuclear Fuel to be Accepted

This implementation alternative involves choosing to accept and manage one of three subalternative amounts of foreign research reactor spent nuclear fuel:

1a. Accept spent nuclear fuel (HEU and/or LEU) only from developing countries. The foreign research reactor spent nuclear fuel from these countries contains approximately 1.9 MTHM, representing 5,000 individual elements, with a volume of 13 m^3 (500 ft^3).

1b. Accept only HEU from the research reactors eligible under the proposed action. The amount of this HEU would be approximately 4.6 MTHM, representing 11,200 elements, with a volume of 61 m^3 (2,250 ft^3).

1c. In addition to foreign research reactor spent nuclear fuel, accept HEU and LEU target materials that were used in Canada, Belgium, Argentina, and Indonesia for the production of medical isotopes. Isotope production targets[3] are irradiated in research reactors and dissolved in acid or base to extract radioisotopes that are used in medical imaging applications. The residual materials after dissolution and extraction of the radioisotopes are referred to here as target material. It is expected that this target material would contain about 0.6 MTHM, representing the uranium content of approximately 620 typical foreign research reactor spent nuclear fuel elements.

Under the last subalternative, HEU target material would be accepted until a suitable LEU target is available. After such a time, target material would be accepted from a foreign research reactor only if that foreign research reactor agrees to convert to use of LEU target.

2.2.2.2 Implementation Alternative 2 - Alternative Policy Durations

The basic implementation of Management Alternative 1 has a duration of 10 years. Two policy duration subalternatives were assessed. These are:

2a. *Five-Year Policy:*

- For foreign research reactors operating on LEU fuel or in the process of converting to LEU fuel when the policy becomes effective, spent nuclear fuel (HEU and LEU) currently stored or generated during the 5-year policy period would be accepted.
- For foreign research reactors operating on HEU fuel when the policy becomes effective and agreeing to convert to LEU fuel, or having life time cores, or planning to shut down by a specific date while the policy is in effect, or for which a suitable LEU fuel is not available, HEU fuel currently stored or generated during the 5-year policy period would be accepted.
- For foreign research reactors that are already shut down, spent nuclear fuel (HEU and LEU) currently stored would be accepted.

3 *Canada, Argentina, and Belgium currently use aluminum-based targets containing HEU, and Indonesia currently uses a target that consists of a layer of HEU oxide material plated on the interior surface of a stainless steel tube.*

The amount of spent nuclear fuel estimated to be accepted for a 5-year policy period under this subalternative is up to approximately 18,800 individual elements containing approximately 13 MTHM, with a volume of 87 m^3 (3,300 ft^3). The distribution by fuel type and geography is as follows:

- By Fuel Type: Aluminum-based — approximately 14,100 elements; 12 MTHM, 83 m^3 (3,100 ft^3).

  TRIGA — approximately 4,700 elements, 1.0 MTHM, 4 m^3 (200 ft^3).

- By Geography: Eastern — approximately 13,400 elements, 9.5 MTHM, 65 m^3 (2,400 ft^3)

  Western — approximately 5,400 elements, 3.4 MTHM, 22 m^3 (900 ft^3)

This subalternative would allow shipments and receipt of foreign research reactor spent nuclear fuel to be made for 8 years starting from the effective date of the policy implementation, as long as the fuel had been generated within the 5-year policy period. The additional 3 years would allow for a cooldown time of fuel discharged late in the 5-year period, the logistics in arranging shipment of this fuel, as well as other possible delays from strikes, court actions, and mechanical problems.

2b. *Indefinite HEU/10-Year LEU Policy:*

- For foreign research reactors operating on LEU fuel or in the process of converting to LEU fuel when the policy becomes effective, LEU spent nuclear fuel currently stored or generated in the 10-year policy period would be accepted within the time period allowed in the basic implementation (13 years). Acceptance of HEU spent nuclear fuel that had been or would be discharged from the reactor would continue indefinitely, until all such HEU spent nuclear fuel had been accepted.

- For foreign research reactors operating on HEU fuel when the policy becomes effective and agreeing to convert to LEU fuel, or planning to shut down by a specific date within 10 years of the effective date of the policy, LEU spent nuclear fuel generated in the 10-year policy period would be accepted within the time period allowed in the basic implementation (13 years). Acceptance of HEU spent nuclear fuel would continue indefinitely, until all such HEU spent nuclear fuel had been received.

- For foreign research reactors operating on HEU fuel when the policy becomes effective and for which a suitable LEU fuel is not available, or having life time cores, HEU spent nuclear fuel would be accepted:

  - from foreign research reactors with lifetime cores, until all such HEU spent nuclear fuel had been accepted, and

  - from other foreign research reactors until all HEU spent nuclear fuel at the reactor on the date the policy becomes effective, or generated within 5 years of that date, had been accepted.

- For foreign research reactors that are already shut down, the LEU spent nuclear fuel would be accepted within the period allowed in the basic implementation. The HEU spent nuclear fuel would be accepted indefinitely, until all such HEU spent nuclear fuel had been accepted.

Under this implementation subalternative, the total amount of foreign research reactor spent nuclear fuel that would be accepted is the same as in the basic implementation of Management Alternative 1.

2.2.2.3 Implementation Alternative 3 - Alternative Financing Arrangements

Under the basic implementation, the costs of participation would be fully subsidized by the United States for developing countries, however, developed countries would be charged a competitive fee.

For this implementation alternative, the cost impacts of the following subalternatives arrangements were evaluated:

- Subsidize all countries;
- Charge all countries the full cost of accepting and managing the foreign research reactor spent nuclear fuel (a full-cost recovery fee); and
- Subsidize developing countries as in the basic implementation, and charge developed countries a full-cost recovery fee.

A full-cost recovery fee would be based on the estimated cost to the United States for the safe, final disposition of the spent nuclear fuel within the United States. This fee could be based on: (1) the cost of chemically separating spent nuclear fuel and disposal of vitrified high-level waste, or (2) interim storage of the spent nuclear fuel followed by direct ultimate disposition.

In theory, this arrangement would cost the United States nothing. All costs would be borne by the foreign nations. However, many developing, and some developed nations probably would decline to pay these high costs, which could lead to HEU spent nuclear fuel stockpiled around the world, much of it remaining in the countries least able to protect it. For many countries, this arrangement would have the same impact as the No Action Alternative.

The amount of spent nuclear fuel that would be accepted under this implementation alternative would be the same as that identified under the basic implementation, or under the implementation subalternatives discussed in Section 2.2.2.1. The actual amount of spent nuclear fuel received could be less than that identified under the basic implementation because, as stated above, some countries may consider the higher fee to be a disincentive. The analysis of environmental impacts for this EIS, however, considered the amount of spent nuclear fuel to be received for this implementation alternative to be unchanged for use as an upper bounding case.

2.2.2.4 Implementation Alternative 4 - Alternative Locations for Taking Title

In the basic implementation, DOE would take title to the foreign research reactor spent nuclear fuel at the limit of U.S. territorial waters (19 km or 12 mi), or the continental U.S. border for shipments from Canada. The location for taking title is relevant to questions of liability and regulatory authority. For example, if DOE were to take title at the foreign research reactor site, there could be additional regulatory burdens on DOE, due to the laws of a particular country being imposed upon the owner of the spent nuclear fuel. The taking of title prior to shipment might impose upon DOE additional legal liability for damages not associated with a nuclear incident covered by the Price-Anderson Act. DOE and the Department of State considered the following three subalternative approaches regarding the locations for taking title to the foreign research reactor spent nuclear fuel:

- Taking title to the foreign research reactor spent nuclear fuel before shipment,
- Taking title at the port(s) of entry, and
- Taking title at the management site(s).

The amount of spent nuclear fuel that would be accepted under this implementation alternative would be the same as that identified under the basic implementation or under the implementation subalternatives discussed in Section 2.2.2.1.

2.2.2.5 Implementation Alternative 5 - Wet Storage Technology for New Construction

Under the basic implementation, storage requiring new construction would employ dry storage technology. As an implementation alternative, DOE has assessed the use of wet storage technologies for new construction, which use water-filled pools to store spent nuclear fuel. Wet storage methods have been used historically at DOE sites and by the nuclear industry.

The amount of spent nuclear fuel that would be accepted under this implementation alternative would be the same as that identified under the basic implementation or under the implementation subalternatives discussed in Section 2.2.2.1.

2.2.2.6 Implementation Alternative 6 - Near Term Conventional Chemical Separation in the United States

Under this implementation alternative, near term conventional chemical separation would be conducted at either the Savannah River Site or the Idaho National Engineering Laboratory. There are both advantages and disadvantages to chemical separation of foreign research reactor spent nuclear fuel. The advantages include the following:

- The high-level radioactive waste from the foreign research reactor spent nuclear fuel would be transformed into forms that are more suitable (i.e., more compact and stable) for storage than intact aluminum-based spent nuclear fuel.
- The high-level waste would be converted to a form that is expected to be acceptable for disposal in a geologic repository.
- Construction of some or all of the new spent nuclear fuel storage space would be avoided.
- The conventional chemical separation facilities already exist, as well as the waste treatment facilities required to put the high-level radioactive and other wastestreams in forms suitable for disposal. In contrast, there are the large technical, cost, and regulatory uncertainties associated with direct disposal of intact foreign research reactor spent nuclear fuel (much of it containing HEU).
- If disposal of intact spent nuclear fuel is shown to be technically infeasible, or if the waste acceptance criteria for a geologic repository require significant dilution of the HEU due to criticality concerns, DOE estimates that the life-cycle costs of chemical separation may be substantially lower than the cost of storage and geologic disposal of intact spent nuclear fuel. (Alternatively, if direct disposal of intact foreign research reactor spent nuclear fuel,

including that containing HEU, is shown to be technically feasible, DOE estimates that the costs of chemical separation and the storage/direct disposal option would be nearly the same.)

The disadvantages include the following:

- Chemical separation would increase the total volume of the waste (including liquid high-level waste raffinates, transuranic wastes, various solid and liquid low-level wastestreams, acidic wastes, chelating and complexing agents, and solvents). Volume reduction and other treatments would be used to prepare these wastes for disposal. (Because the requirements for direct disposal of aluminum-based spent nuclear fuel have not been established, the character and volumes of waste associated with direct disposal are uncertain.)

- The separated uranium, which DOE would prefer to blend down to LEU, would have to be stored until it could be sold or otherwise disposed of.

- The forms of the wastes generated by chemical separation are complex, involving corrosive, flammable and toxic liquids.

- The use of chemical separation by the United States as a spent nuclear fuel management technology could increase the accumulation of stockpiles of HEU unless the HEU is blended down. The United States does not engage in chemical separation for nuclear explosive purposes, and seeks to eliminate, where possible, the accumulation of stockpiles of HEU or plutonium. The United States nuclear weapons nonproliferation policy on reprocessing is summarized in the White House Fact Sheet on Nonproliferation and Export Control Policy dated September 27, 1993. A copy of this policy is included in Appendix G of this EIS.

Taking these advantages and disadvantages into account, chemical separation of foreign research reactor spent nuclear fuel in existing facilities is not preferred by DOE as a technology for routine management of spent nuclear fuel in the United States. Nonetheless, chemical separation remains a reasonable alternative in light of DOE's substantial technical experience in these operations and the availability of existing facilities.

DOE is considering the development and use of various alternatives to chemical separation for foreign research reactor spent nuclear fuel stabilization, interim storage and conditioning for disposal under Management Alternative 1, Implementation Alternative 7 in this EIS. This initiative is discussed in more detail in a DOE memorandum of December 28, 1994 from Thomas P. Grumbly to Jill E. Lytle (see Appendix G). In fact, development, demonstration and implementation of a new treatment and/or packaging technology is a key element of the preferred alternative as described in Section 2.9.

The Nuclear Waste Policy Act of 1982 (as amended) authorizes disposal of the foreign research reactor spent nuclear fuel in a geologic repository (if DOE takes title to such fuel). The foreign research reactor spent nuclear fuel, and/or the high-level radioactive waste that would result from chemical separation of the foreign research reactor spent nuclear fuel, would require storage until its disposal in such a geologic repository. A determination of whether the foreign research reactor spent nuclear fuel can be safely disposed of in a geologic repository will depend on the outcome of scientific analyses, including a repository performance analysis considering the final form in which the foreign research reactor spent nuclear fuel would be emplaced in the repository.

Near term conventional chemical separation of foreign research reactor spent nuclear fuel at the other three potential foreign research reactor spent nuclear fuel management sites was not analyzed because the Oak Ridge Reservation and the Nevada Test Site do not have facilities in which such chemical separation could be conducted, and the facilities at the Hanford Site are no longer operable. To consider chemical separation at any of these three sites, the foreign research reactor spent nuclear fuel would need to be stored in the United States during the period of time that a new chemical separation facility at one of these sites was designed, a project-specific NEPA review was conducted, and the facility constructed and put into operation. Such activities could not be completed in the near term and, accordingly, these sites were not considered reasonable alternatives for near term chemical separation.

Solid low-level radioactive waste and wastewater generated by chemical separation would be managed in the same manner as the similar wastes generated by the storage of the intact foreign research reactor spent nuclear fuel. Discussion of waste generation from storage is included in Appendix F, Section F.4. Chemical separation would also generate five types of waste that would not result from storage of intact foreign research reactor spent nuclear fuel: high-level radioactive waste, hazardous waste, mixed hazardous and radioactive waste, and low-level "saltstone" waste.

Following chemical separation of the foreign research reactor spent nuclear fuel, the resulting high-level radioactive wastes would be managed along with substantial existing inventories of identical waste. Management of the high-level radioactive wastes would include the following:

1. The high-level wastes would be transferred to storage tanks and kept there pending processing;
2. The wastes would be pretreated in preparation for further processing;
3. To preclude the necessity for transporting liquid high-level wastes, these wastes would be processed on the sites where they were generated:
   a. At the Savannah River Site, the wastes would be:
      1) Vitrified in the Defense Waste Processing Facility;
      2) The borosilicate glass resulting from vitrification would be stored pending disposal;
   b. At the Idaho National Engineering Laboratory, the wastes would be:
      1) Calcined to produce a more easily stored waste form;
      2) Stored in the calcine form pending development of a process and facility for final processing;
      3) After further research and development regarding conversion techniques and waste forms, the calcine would be converted to a form suitable for geologic disposal and stored pending disposal;
4. The final waste form would be transported to and disposed of in a geologic repository.

Transuranic wastes[4] would be stored on the site where the chemical separation would be accomplished until a permanent disposal facility, such as the Waste Isolation Pilot Plant, becomes available. Site treatment plans for low-level and transuranic mixed wastes are now being developed. Hazardous wastes would be sent to a licensed commercial treatment, storage and disposal facility. Saltstone, a mixture of low-level waste and concrete that is a by-product of high-level radioactive waste vitrification at the Savannah River Site's Defense Waste Processing Facility, would be pumped into above-ground concrete vaults onsite, where it would harden into a concrete monolith.

The Savannah River Site currently has chemical separation facilities. This capability, however, is limited to aluminum-based spent nuclear fuel. In contrast, the Idaho National Engineering Laboratory has facilities that can chemically separate both aluminum-based and TRIGA foreign research reactor spent nuclear fuel. However, these facilities would require some upgrades in order to accomplish this chemical separation. The existing dissolvers and calcination vessel could be used at the start of chemical separation activities, but would have to be replaced within a few years. A new tank farm and set of calcine bins would have to be built. Furthermore, this site does not have an existing vitrification facility or a glass waste storage building, as the Savannah River Site does. Upgrading the facilities at the Idaho National Engineering Laboratory would require additional time and funding. This EIS analyzes the impacts of chemically separating aluminum-based foreign research reactor spent nuclear fuel at both the Savannah River Site and the Idaho National Engineering Laboratory, but considers chemical separation of TRIGA foreign research reactor spent nuclear fuel only at the Idaho National Engineering Laboratory.

Under the near term chemical separation alternative, there are two components: Extent of the Chemical Separation and Uranium Disposition. Each of these components is discussed below.

Extent of the Chemical Separation

At each of the two potential sites, the foreign research reactor spent nuclear fuel could be chemically separated in a dedicated mode or as part of larger scale chemical separation activities.

Chemical Separation at the Savannah River Site Dedicated to Foreign Research Reactor Spent Nuclear Fuel: DOE would chemically separate all 18.2 MTHM of the aluminum-based foreign research reactor spent nuclear fuel, shown previously in Table 2-1. The Savannah River Site has facilities that could perform the chemical separation, so no new chemical separation facilities would need to be constructed.

DOE and the Department of State have included in this EIS analysis all 18.2 MTHM of aluminum-based foreign research reactor spent nuclear fuel in Table 2-1 because this is the maximum that could be considered. It is not possible to specify how much of the aluminum-based foreign research reactor spent nuclear fuel might be chemically separated before the chemical separation facilities would have been shut down, so the analysis is based on the entire amount. If chemical separation of all the foreign research reactor spent nuclear fuel were selected, existing chemical separation facilities would be required to remain operational for the 13-year duration of receipts. Maintenance and operation of a facility dedicated solely to chemical separation of foreign research reactor spent nuclear fuel is considered to be an inefficient use of such facilities. Thus, this is a nonpreferred subalternative.

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4 *No transuranic waste would be generated if the transuranic elements (mostly plutonium) were not extracted during chemical separation. The trace amounts of these elements that exist in the foreign research reactor spent nuclear fuel could remain in the high-level wastestream.*

Chemical Separation at the Idaho National Engineering Laboratory Dedicated to Foreign Research Reactor Spent Nuclear Fuel: DOE would restart the facilities and chemically separate all the aluminum-based and TRIGA foreign research reactor spent nuclear fuel, shown previously in Tables 2-1 and 2-2. The Idaho National Engineering Laboratory does not have all the facilities required to perform the chemical separation, so some new facilities would need to be constructed. Furthermore, DOE announced a Record of Decision on May 30, 1995 for the Programmatic SNF&INEL Final EIS. Chemical separation at Idaho National Engineering Laboratory was not included in this Record of Decision, so additional site-specific NEPA documentation would be required to restart these chemical separation facilities.

DOE and the Department of State have included in this analysis all 19.2 MTHM of foreign research reactor spent nuclear fuel in Tables 2-1 and 2-2, because this is the maximum that could be considered. It is not possible to specify how much of the foreign research reactor spent nuclear fuel might be chemically separated in this case, so the analysis is based on the entire amount. Up to approximately 12 years of operation would be required to chemically separate this amount. The construction and operation of new facilities for chemical separation dedicated to foreign research reactor spent nuclear fuel is considered inefficient, and therefore, this subalternative is not preferred.

Chemical Separation at the Savannah River Site as Part of Larger Scale Activities: DOE is in the process of preparing other NEPA reviews and making decisions that could affect the decisions to be made in this EIS. The Interim Management of Nuclear Materials Final EIS (DOE, 1995a) analyzed alternatives for stabilization of nuclear materials currently stored at the Savannah River Site that represent health and safety risks, as stored in their current forms and locations. The nuclear materials in the Interim Management of Nuclear Materials Final EIS that most closely resemble the aluminum-based foreign research reactor spent nuclear fuel are the Mark-16 and Mark-22 fuels. The preferred alternative for these fuels, as announced in the *Federal Register* (60 FR 65300), is chemical separation. Therefore, the near term chemical separation of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site as part of larger scale activities is predicated upon a decision by DOE to use the chemical separation facilities at the Savannah River Site to chemically separate the Mark-16 and Mark-22 fuels under the Interim Management of Nuclear Materials Final EIS.

The Programmatic SNF&INEL Final EIS (DOE, 1995c) considered the alternative site(s) where DOE's spent nuclear fuel (including foreign research reactor spent nuclear fuel) would be managed. DOE announced in its Record of Decision on May 30, 1995 that it intends to consolidate all its aluminum-based spent nuclear fuel at the Savannah River Site. DOE could also chemically separate other aluminum-based spent nuclear fuel that is transported to the Savannah River Site under this EIS.

The aluminum-based foreign research reactor spent nuclear fuel shown in Table 2-1 would be chemically separated along with other DOE aluminum-based spent nuclear fuel selected for chemical separation. This could amount to a maximum of 18.2 MTHM of aluminum-based foreign research reactor spent nuclear fuel, 0.56 MTHM of target material under Implementation Subalternative 1c (Section 2.2.2.1), approximately 28.8 MTHM of other aluminum-based spent nuclear fuel currently stored at the Savannah River Site (Wichmann, 1995), and approximately 3.4 MTHM of aluminum-based spent nuclear fuel that could be transported to the Savannah River Site (Wichmann, 1995). In all, about 51 MTHM could be chemically separated at the Savannah River Site, requiring up to approximately 13 years of operation. The environmental impacts analysis is presented in Section 4.3.6.

Chemical Separation at the Idaho National Engineering Laboratory as Part of Larger Scale Activities: Volume 2 of the Programmatic SNF&INEL Final EIS (DOE, 1995c) includes a brief analysis of the alternative of restarting the chemical separation facilities at the Idaho National Engineering Laboratory for stabilization of nuclear materials. These facilities are currently being cleaned up in preparation for decommissioning, so restarting them would require additional site-specific NEPA documentation, but DOE is not currently performing NEPA analysis on restarting these facilities.

Volume 1 of the Programmatic SNF&INEL Final EIS (DOE, 1995c) considered the alternative site(s) where DOE's spent nuclear fuel (including foreign research reactor spent nuclear fuel) would be managed. DOE could chemically separate other spent nuclear fuel that is transported to the Idaho National Engineering Laboratory under this EIS.

All the aluminum-based spent nuclear fuel would be chemically separated along with all the TRIGA foreign research reactor spent nuclear fuel and certain other spent nuclear fuel from the Idaho National Engineering Laboratory. The amount of aluminum-based spent nuclear fuel would be approximately 51 MTHM, as described above. The additional spent nuclear fuel would include 1.0 MTHM of TRIGA foreign research reactor spent nuclear fuel and approximately 13 MTHM of TRIGA and zirconium-based spent nuclear fuel from onsite facilities (Cottam, 1995). In all, about 65 MTHM could be chemically separated under this program at Idaho National Engineering Laboratory, requiring up to approximately 12 years of operation. The environmental impact analysis is presented in Section 4.3.6.

Uranium Disposition

Chemical separation, as the name implies, would separate the uranium from the waste products. The separated LEU could be sold to the commercial sector for reuse as reactor fuel. The HEU disposition issue is being considered in a separate DOE NEPA document, the Disposition of Surplus Highly Enriched Uranium EIS. If DOE decides to blend down the HEU that is within the scope of the Disposition of Surplus Highly Enriched Uranium EIS, then the HEU that would be recovered in this implementation alternative would also be blended down. Conversely, if DOE decides not to blend down the HEU that is within the scope of the Disposition of Surplus Highly Enriched Uranium EIS, then the HEU that would be recovered in this implementation alternative would also not be blended down. Until this decision is made, however, DOE may decide to blend down HEU in specific instances. For example, DOE recently announced its decision (60 FR 65300), to blend down the HEU solutions at the Savannah River Site under the Interim Management of Nuclear Materials Final EIS.

The options for HEU disposition at the Savannah River Site are:

1. Blending it down to less than 20 percent enrichment inside the chemical separation facilities;

2. Blending it down to less than two percent enrichment inside the chemical separation facilities and then processing it to an oxide in the existing FA-Line at the Savannah River Site; and

3. Completing construction of the Uranium Solidification Facility at the Savannah River Site, then processing the HEU directly to an oxide, followed by storage in a safe, secure facility.

The options for HEU disposition at Idaho National Engineering Laboratory are:

1. Blending it down to less than 20 percent inside the chemical separation facilities, and

2. Processing the HEU directly to an oxide.

Some minor modifications to the facility would be necessary to blend down the HEU. No modifications would be necessary to process it directly to an oxide. For either option, additional NEPA documentation would be required.

Under any of the blending down options, the nuclear weapons nonproliferation goal would be satisfied because the material would not be usable in weapons after it was blended down. This material could be returned to the commercial sector and reused in nuclear reactors.

If the HEU were not blended down, DOE and the Department of State might be accused of accepting the foreign research reactor spent nuclear fuel in order to stockpile HEU for future weapons use. To address this concern, DOE and the Department of State would, over time, place the separated HEU under International Atomic Energy Agency safeguards. DOE and the Department of State have identified three possible means of implementing this International Atomic Energy Agency safeguards initiative which would require the use of a finite storage area (Material Balance Area) subject to inspections by the International Atomic Energy Agency. These are:

1. DOE could use the only available Material Balance Area: Vault 16 at the Y-12 Plant on the Oak Ridge Reservation. This vault's capacity is about 40 MTHM of HEU and it presently contains only about 10 MTHM, so the available capacity is about 30 MTHM. Chemical separation of all the spent nuclear fuel in this implementation alternative would recover less than 25 MTHM of HEU, so there is sufficient capacity in Vault 16 to store all this material.

2. A new Material Balance Area could be set up at the Savannah River Site or the Idaho National Engineering Laboratory.

3. The HEU could be stored in existing vaults, and brought out to a temporary Material Balance Area for each International Atomic Energy Agency inspection.

If a decision is made to chemically separate the foreign research reactor spent nuclear fuel, it would be DOE's preference to blend down the HEU to LEU and thus preclude the possibility of this material ever being used in a nuclear weapon.

2.2.2.7 Implementation Alternative 7 - Developmental Treatment and/or Packaging Technologies

Under this implementation alternative, DOE and the Department of State would initiate a development program that could lead to a decision to construct and operate a new facility for treatment and/or packaging of foreign research reactor spent nuclear fuel. The purpose of this potential new facility would be to treat, package and store the foreign research reactor spent nuclear fuel in a manner suitable for geologic disposal, without necessarily separating the fissile materials. Other goals of the development process would be to define a technology and facility that would operate safely, meet or exceed all applicable environmental requirements (including minimization of waste volumes, toxicity, and mobility), and be consistent with U.S. nuclear weapons nonproliferation policies.

There are numerous technologies that DOE and the Department of State could consider under such a development program. These technologies could be applied at any one of the five potential foreign research reactor spent nuclear fuel management sites, and most would require the construction of totally new facilities (although some could be implemented through modifications to existing facilities). The potential environmental impacts of the construction and operation of such a new facility cannot be estimated at this time because the technologies are still developmental and the hypothetical new facility has not been designed. Implementation of any of these technologies would require additional NEPA analysis and documentation, including additional opportunities for public review and comment.

Many of the potentially applicable technologies are already being considered under the DOE Office of Spent Fuel Management Technology Integration Working Group. A development program, such as the one that would be implemented under this alternative, is outlined in the DOE Spent Nuclear Fuel

Technology Integration Plan (DOE, 1994c). A number of these developmental technologies have progressed beyond initial technical feasibility studies and have reasonably defined cost and schedule estimates for further development. Technologies, such as the Plasma Arc Treatment Process and the Electrometallurgical Treatment Process, are being developed by the Pacific Northwest Laboratory and the Argonne National Laboratory, respectively. Other developmental technologies, however, require additional evaluation prior to undergoing detailed development efforts. Some of the development technologies and criticality prevention techniques are briefly described below. Criticality prevention is discussed in Section 2.6.1.

Developmental Treatment Technologies

Chop and Dilute: The foreign research reactor spent nuclear fuel could be rendered into shards in a mechanical chopper and added to shards of depleted uranium-aluminum alloy to prevent a criticality in the repository. The mixture would have an enrichment of no more than one percent (WSRC, 1994a).

Chop and Poison: The foreign research reactor spent nuclear fuel could be rendered into shards in a mechanical chopper and a neutron poison could be added to prevent a criticality in the repository (WSRC, 1994a). A neutron poison is an element that absorbs neutrons without fissioning, thus preventing a fission chain reaction.

Melt and Dilute: The foreign research reactor spent nuclear fuel could be melted with depleted uranium metal added to the molten mixture to prevent a criticality in the repository (WSRC, 1994a).

Melt and Poison: The foreign research reactor spent nuclear fuel could be melted and a neutron poison could be added to the molten mixture to prevent a criticality in the repository (WSRC, 1994a).

Electrometallurgical Treatment: The foreign research reactor spent nuclear fuel could be dissolved, then the aluminum could be separated from the uranium and fission products in an electrorefiner. DOE has proposed to demonstrate this process as a management option for a variety of DOE-owned spent nuclear fuel. This process would produce a mineral waste form containing most of the fission products and a metal alloy containing the rest of the fission products (DOE, 1994c).

Plasma Arc Treatment: The foreign research reactor spent nuclear fuel would be placed directly into a plasma centrifugal furnace with other material (low-enriched uranium, depleted uranium, and neutron absorbers) where it would be melted and converted into a ceramic material.

Chloride Volatility Treatment: The foreign research reactor spent nuclear fuel could be completely volatized to chlorides. This process is being investigated at the Idaho National Engineering Laboratory and would require about 15 years to develop. Then the uranium could be separated and the fission products could be converted to oxides or fluorides for vitrification (DOE, 1994c).

Glass Material Oxidation and Dissolution System: The foreign research reactor spent nuclear fuel could be converted to a glass form in this single-step process. This process was recently invented at Oak Ridge National Laboratory, and has been demonstrated at the laboratory scale. The foreign research reactor spent nuclear fuel would be melted together with glass frit and all process chemistry would occur in this molten mixture. Then, depleted uranium would be added to resolve criticality concerns before the mixture is poured into canisters (DOE, 1994c).

Dissolve and Dilute: The foreign research reactor spent nuclear fuel could be dissolved in acid and depleted uranium could be added to the solution to reduce the enrichment to no more than one percent to prevent a criticality in the repository. Then the solution would be vitrified (WSRC, 1994a).

Dissolve and Poison: The foreign research reactor spent nuclear fuel could be dissolved in acid and a neutron poison could be added to prevent a criticality in the repository. Then the solution would be vitrified (WSRC, 1994a).

Developmental Packaging Technologies

Direct Disposal in Small Packages: This is a variation of the "direct disposal"concept. The foreign research reactor spent nuclear fuel could be packed intact into small waste packages to limit the amount of fissile material in any single package. Neutron poisons would also be packed in the packages in the spaces between the fuel element plates or rods to prevent a criticality in the repository.

Can-in-Canister: The foreign research reactor spent nuclear fuel could be canned, then the cans could be encapsulated in glass inside of canisters. The encapsulation process could be performed in the Defense Waste Processing Facility at the Savannah River Site, using a high-level waste glass. In effect, the cans containing foreign research reactor spent nuclear fuel would displace an equal volume of high-level waste glass inside standard Defense Waste Processing Facility canisters.

2.3 Management Alternative 2 — Facilitate the Management of Foreign Research Reactor Spent Nuclear Fuel Overseas

Under this management alternative, DOE and the Department of State would seek to encourage and facilitate the management of foreign research reactor spent nuclear fuel overseas in a manner that would be consistent with U.S. nuclear weapons nonproliferation policy. DOE and the Department of State have evaluated the following two subalternatives:

1a. Overseas Storage - Encourage and assist foreign research reactors that are able to store their spent nuclear fuel in facilities in their own countries, or in developing countries, as a step toward the final disposition of the spent nuclear fuel. U.S. assistance would be provided to ensure that appropriate storage technologies, regulations, and safeguards were applied.

In some cases, this subalternative might be implemented by expansion of the storage facilities located at the foreign research reactor sites. However, many foreign research reactor operators are associated with academic institutions with limited budgets, or have building restrictions for site-specific reasons (e.g., no physical space for expansion). Thus, the opportunities for expanded spent nuclear fuel storage at foreign research reactor sites may be limited or even nonexistent. In countries with established nuclear power programs, management might also be provided at the sites where such countries will store their power reactor spent nuclear fuel. Some foreign research reactor operators may also be able to make arrangements for indefinite storage at sites in developing countries. Ultimate disposition of the foreign research reactor spent nuclear fuel would still have to be arranged at the conclusion of the management period. In the meantime, foreign research reactor spent nuclear fuel containing HEU would be stored in up to 41 countries around the world.

1b. Overseas Reprocessing - Provide nontechnical (financial and/or logistical) assistance to foreign research reactors and reprocessors to facilitate reprocessing spent nuclear fuel overseas in facilities operated under international safeguards sufficient to satisfy U.S. nuclear weapons nonproliferation concerns. Wherever possible, the wastes resulting from this reprocessing would be returned to the country in which the spent nuclear fuel was irradiated. If the reprocessing wastes cannot be returned to the country in which the spent nuclear fuel was irradiated, such wastes might be accepted by the United States for storage and ultimate geologic disposal.

The advantages and disadvantages of the technology used for reprocessing overseas would be essentially the same as those described for chemical separation in the United States as discussed in Section 2.2.2.6.

The overseas reprocessing option will be evaluated in terms of whatever is supportive of the U.S. nuclear weapons nonproliferation policy on HEU minimization. For example, factors such as the following would have to be considered:

- An expectation that HEU separated during reprocessing would be blended down to LEU for research reactors which are converting to LEU.
- The foreign reprocessors would provide the capability to reprocess LEU as well as HEU.
- Research reactors would be encouraged to convert to LEU if an LEU fuel exists or is developed that will allow such operation.

Arrangements would have to be worked out with foreign reprocessors that would be supportive of U.S. nuclear weapons nonproliferation objectives to minimize the civil use of HEU worldwide.

Reprocessing of spent nuclear fuel is a well-established technology which is based on the same principles as chemical separation in the United States as discussed in Section 2.6.5.2, and an international commercial market has developed with a total annual capacity of several thousand MTHM (BNFL, 1994; Cogema, 1994). Large portions of this capacity are oriented toward commercial spent nuclear fuel. While these facilities are technically capable of reprocessing foreign research reactor spent nuclear fuel with relatively minor modifications [e.g., blending of the foreign research reactor spent nuclear fuel in the dissolver(s) with depleted uranium], for contractual, economic, and schedule considerations, these commercial spent nuclear fuel reprocessing facilities are less inclined to consider the potential foreign research reactor spent nuclear fuel market.

Several large (hundreds of MTHM per year) spent nuclear fuel reprocessing facilities exist for LEU spent nuclear fuel, and the annual capacity of existing facilities for research and HEU spent nuclear fuel is in the tens of MTHM range. The British facilities at Dounreay are currently capable of reprocessing foreign research reactor spent nuclear fuel. The French facilities at Marcoule are planning reprocessing of French research reactor spent nuclear fuel in the near future, and the Dounreay facility is pursuing additional contracts with foreign research reactor operators for reprocessing their spent nuclear fuel. The estimated schedule of foreign research reactor spent nuclear fuel shipments corresponds to a maximum of 2 MTHM per year, which could be accommodated by the existing reprocessing capacity for these fuels. Significantly, the most likely candidate facilities for reprocessing foreign research reactor spent nuclear fuel are located in Europe and are operated under Euratom and International Atomic Energy Agency safeguards. These facilities also offer full spent nuclear fuel management capabilities, including spent nuclear fuel storage prior to reprocessing, solidification of wastes, product and waste transportation, and assay adjustments (i.e., blending of HEU to LEU). Arrangements for shipment and disposition of the processing wastes would have to be implemented. Most processing contracts with European facilities require return of the wastes to the generator (in this case, the foreign research reactor operators) of the spent nuclear fuel.

DOE has considered the possibility of accepting, in the United States, the vitrified waste from the reprocessing of foreign research reactor spent nuclear fuel overseas. The environmental impacts from the receipt and storage of the waste in the United States are presented in Section 4.4.2.

2.4 Management Alternative 3 - Combination of Elements from Management Alternatives 1 and 2 (Hybrid Alternative)

In implementing the proposed action, DOE and the Department of State could combine implementation elements from the management alternatives analyzed in Sections 2.2 and 2.3. For example, DOE and the Department of State could consider partial storage or reprocessing overseas and partial storage or chemical separation in the United States. The impacts to the U.S. environment from hybrid alternatives would be bounded by the analysis presented in this EIS for each of the implementation alternatives for Management Alternative 1, because for each implementation alternative, the analysis considers the maximum amount of spent nuclear fuel that could be accepted, stored, and/or chemically separated in the United States.

For the purpose of illustration, DOE and the Department of State have considered an example of a hybrid alternative which is a combination of implementation elements of Management Alternatives 1 and 2. This hybrid alternative is described below.

Under this hybrid alternative, DOE and the Department of State would provide nontechnical (financial and/or logistical) assistance to foreign research reactor operators and reprocessors to facilitate the reprocessing of any foreign research reactor spent nuclear fuel at western European reprocessing facilities (i.e., Dounreay or Marcoule), as in Management Alternative 2, for foreign research reactors in countries that could accept back the reprocessed waste; and DOE and the Department of State would accept and manage the rest of foreign research reactor spent nuclear fuel in the United States, as in Management Alternative 1.

In order to comply with U.S. nuclear weapons nonproliferation policy, bilateral agreements would need to be established with the foreign governments involved to ensure that the conditions discussed in Section 2.3 for overseas reprocessing (Management Alternative 2, Implementation Alternative 1b) would be met before DOE and the Department of State would consider implementation of this hybrid alternative.

Based on the current capabilities of the reprocessors overseas, the spent nuclear fuel to be considered for reprocessing would be aluminum-based. TRIGA spent nuclear fuel could also be considered if such capability is developed; however, for the purposes of the analysis, TRIGA spent nuclear fuel would be assumed to be accepted in the United States for storage. Table 2-3 lists the countries that may be able to accept the reprocessing waste and the amount of spent nuclear fuel to be considered for reprocessing overseas. Table 2-4 shows the amount of spent nuclear fuel that would be accepted in the United States.

Under this hybrid alternative, the aluminum-based foreign research reactor spent nuclear fuel to be accepted in the United States would be chemically separated at the Savannah River Site as in Implementation Alternative 6, which is discussed in Section 2.2.2.6. The TRIGA spent nuclear fuel would be transported to the Idaho National Engineering Laboratory, where it would be stored at existing storage facilities until ultimate disposition. The distribution of the spent nuclear fuel considered in this hybrid alternative is consistent with the Programmatic SNF&INEL Final EIS (DOE, 1995c) Regionalization by Fuel Type alternative.

All the other components of this hybrid alternative are the same as the basic implementation of Management Alternative 1, specifically:

- a policy duration of 10 years with a period of acceptance of spent nuclear fuel in the United States of 13 years;

Table 2-3 Spent Nuclear Fuel Considered for Reprocessing Overseas (Hybrid Alternative Example)

Country	*Number of Elements*	*Mass (MTHM)*[a]
Belgium	1,766	0.730
France	1,962	3.442
Germany	1,504	0.909
Italy	150	0.043
Spain	40	0.016
Switzerland	159	0.128
United Kingdom	12	0.004
Total	**5,593**	**5.272**

[a] *To derive mass in kilograms, multiply by 1,000.*

Table 2-4 Amount and Distribution of Foreign Research Reactor Spent Nuclear Fuel to be Accepted in the United States (Hybrid Alternative Example)

Type	*Number of Elements*	*Mass (MTHM)*[a]	*Number of Shipments*
Aluminum-Based	12,210	12.912	406
Eastern[b]	7,593	8.647	275
Western[b]	4,617	4.263	131
TRIGA	4,940	1.033	162
Eastern[b]	3,245	0.528	107
Western[b]	1,695	0.505	55
Total	**17,150**	**13.945**	**568**

[a] *To derive mass in kilograms, multiply by 1,000.*

[b] *Refers to the location of the likely port(s) of entry to the United States.*

- a financing arrangement by which the United States would bear the full cost for transporting and managing the foreign research reactor spent nuclear fuel received from developing countries, but would charge developed countries (if any) a competitive fee;
- taking title to the foreign research reactor spent nuclear fuel at the U.S. territorial waters limit or continental U.S. borders for shipments from Canada;
- marine transport of the foreign research reactor spent nuclear fuel by chartered and/or regularly scheduled commercial ships;
- ports of entry that qualify on the bases of criteria discussed in this EIS; and
- ground transport from ports of entry to the Savannah River Site and Idaho National Engineering Laboratory by truck, rail, or barge, or a combination of these modes.

The impacts to the U.S. environment from this hybrid alternative would be bounded by the Savannah River Site portion of Implementation Alternative 6 (Near Term Chemical Separation in the United States at the Savannah River Site), which considers the acceptance of approximately 22,700 elements of foreign research reactor spent nuclear fuel in the United States versus approximately 17,100 (13.9 MTHM) elements in this hybrid alternative; the chemical separation of approximately 17,800 aluminum-based elements versus approximately 12,200 aluminum-based elements in this hybrid alternative; and the storage of approximately the same number of TRIGA elements as under the basic implementation.

The environmental impacts and policy considerations of the hybrid alternative are discussed in Section 4.5.

2.5 No Action Alternative

In the No Action Alternative, the United States would neither manage foreign research reactor spent nuclear fuel containing uranium enriched in the United States, nor provide technical assistance or financial incentives for overseas storage or reprocessing. In this case, no foreign research reactor spent nuclear fuel shipments to the United States and no assistance to foreign countries for managing foreign research reactor spent nuclear fuel overseas would take place. The No Action Alternative would have environmental impacts outside the United States which are not in the scope of this EIS. Policy considerations are discussed in Section 4.6.

2.6 Characteristics of the Components of the Basic Implementation

This section summarizes information on the selection process for some of the components of the basic implementation, as well as characteristics, assumptions and physical parameters used in the environmental impact analysis. This section provides characteristics of the spent nuclear fuel to be received, the types of transportation casks considered, the marine ports considered and method of their identification, the ground transportation routes considered, method of identification, typical characteristics of the wet and dry storage technologies, descriptions of the designs of facilities for both dry and wet storage technologies, descriptions of chemical separation facilities, and details on the site-specific options in managing the foreign research reactor spent nuclear fuel.

2.6.1 Characteristics and Types of Foreign Research Reactor Spent Nuclear Fuel

Spent nuclear fuel is fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated. Fuel in a reactor consists of fuel elements that come in many configurations; but generally consist of the fuel matrix, cladding and structural parts. The matrix, which contains the fissionable material is typically in the form of plates or cylindrical pellets. The cladding (typically zirconium, aluminum, or stainless steel) surrounds the fuel, confining and protecting it. Structural parts (generally nickel alloys, stainless steel, zirconium, or aluminum) hold the fuel elements in the proper configuration in the reactor core.

Spent nuclear fuel is radioactive because of the presence of the radioactive isotopes, which are products of the fission process. The radiation of most concern from spent nuclear fuel is gamma rays. Although the radiation levels can be very high, the gamma ray intensities are readily reduced by shielding the spent nuclear fuel elements with such materials as steel, lead, concrete, and water during the various management phases of handling, transporting, or storing the spent nuclear fuel elements.

An issue associated with the management of spent nuclear fuel containing significant amounts of fissionable material is the potential for a self-sustaining nuclear fission process called criticality. Prevention of criticality conditions enters in the design of the spent nuclear fuel transportation casks, the spent nuclear fuel storage and processing facilities, and the spent nuclear fuel packaging for ultimate disposition. In general, criticality prevention is accomplished by either controlling the amount of fissionable material present within a certain volume (dilution or spatial separation techniques) or by introducing neutron absorbing materials that reduce the number of neutrons available to the fission process (poisoning technique). The criticality issue has been addressed in all implementation alternatives considered for the management of the foreign research reactor spent nuclear fuel.

Two types of foreign research reactor spent nuclear fuel are covered under the proposed action. They are aluminum-based fuel and TRIGA reactor type fuel. The aluminum-based fuel refers to fuels that consist of an alloy of uranium and aluminum, or a dispersion of uranium-bearing compound in aluminum, both clad in aluminum. The enrichment of uranium can be either HEU or LEU. Details on the physical and nuclear characteristics of the aluminum-based foreign research reactor spent nuclear fuel can be found in Appendix B. The aluminum-based fuels are used in various reactor types in different forms and geometries. The spent nuclear fuel element geometries are either cylindrical, boxed type, annular with hundreds of involute plates, or pin cluster. The aluminum-based fuel forms are either plates, tubes, rods, or pins. The ^{235}U[5] content of a fuel element prior to irradiation in a reactor (i.e., fresh fuel element) can vary from about 3 g (.11 oz) to about 8,500 g (300 oz). The length of an individual element can vary from 22 cm (8.7 in) to about 300 cm (118 in).

The TRIGA reactor fuel uses uranium-zirconium hydride (U-Zr-H_x) fuel material in which the hydrogen moderator is homogeneously contained within the fuel material. The initial ^{235}U content of each rod varies between 38 g (1.3 oz) and 133 g (4.7 oz). The overall length of a TRIGA fuel rod is approximately 76 cm (30 in), and the weight is between approximately 1 kg (2.2 lb) and about 4 kg (8.8 lb). Details on the physical and nuclear characteristics of the TRIGA foreign research reactor spent nuclear fuel can be found in Appendix B.

In contrast, a typical nuclear power reactor fuel (e.g., pressurized water reactor fuel) is three to five percent enriched uranium-oxide. The fuel form is ceramic pellets combined into rods, and the cladding is zircaloy or stainless steel. A typical pressurized water reactor fuel assembly weighs 682 kg (1,500 lb) and has a length of 389 cm (13 ft). Figure 2-5 graphically depicts the differences in size of a typical pressurized water reactor assembly, a typical aluminum-based fuel element, and a TRIGA fuel element. Additional detailed information on the aluminum-based and TRIGA fuels are provided in Appendix B of this EIS.

In addition to aluminum-based and TRIGA-type spent nuclear fuel, target material containing HEU is considered for management under Implementation Alternative 1, subalternative 1c (Section 2.2.2.1). Targets are irradiated in a research reactor to produce molybdenum-99, a medical isotope. Molybdenum production peaks at a low burnup, about three percent. Once the target is removed from the reactor, the fuel is dissolved in acid, and molybdenum-99 is separated from the solution. The residual material after removal of molybdenum-99 is called target material, and is currently kept in solution form. The target material considered for management would be put in U_3O_8 or UO_2 form and canned for transport to the United States. It is expected that the target material would contain about 0.6 MTHM (the uranium content of 620 typical Material Test Reactor [MTR] elements) and a volume of 6.5 m^3 (229.5 ft^3). This material could be brought to the United States in cans having a cavity of 6.4 cm (2.5 in) in diameter and 28 cm (11 in) long, and containing between 40 g to 200 g (1.41 oz to 7 oz) of ^{235}U each.

2.6.2 Transportation Casks

Spent nuclear fuel elements are transported in stainless steel packages called transportation casks, or just casks. A typical cask for the transportation of foreign research reactor spent nuclear fuel elements is shown in Figure 2-6. Detailed descriptions of typical casks are provided in Appendix B (Section B.2).

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5 *During the enrichment process, the amount of fissionable Uranium-235 (^{235}U) is increased. Uranium increased to less than 20 percent ^{235}U is called LEU. Uranium enriched to 20 percent or greater ^{235}U is called HEU.*

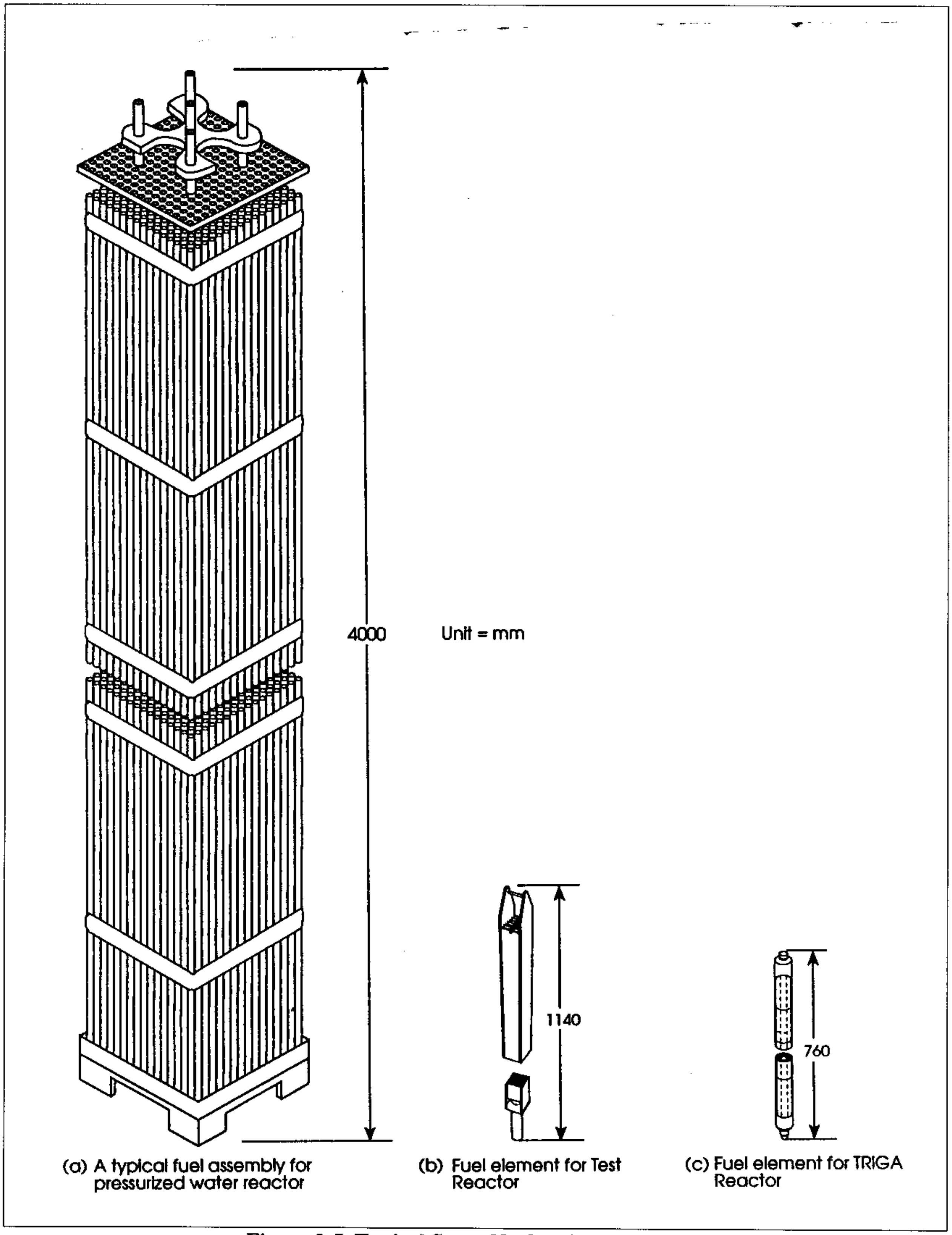


Figure 2-5 Typical Spent Nuclear Fuel Elements

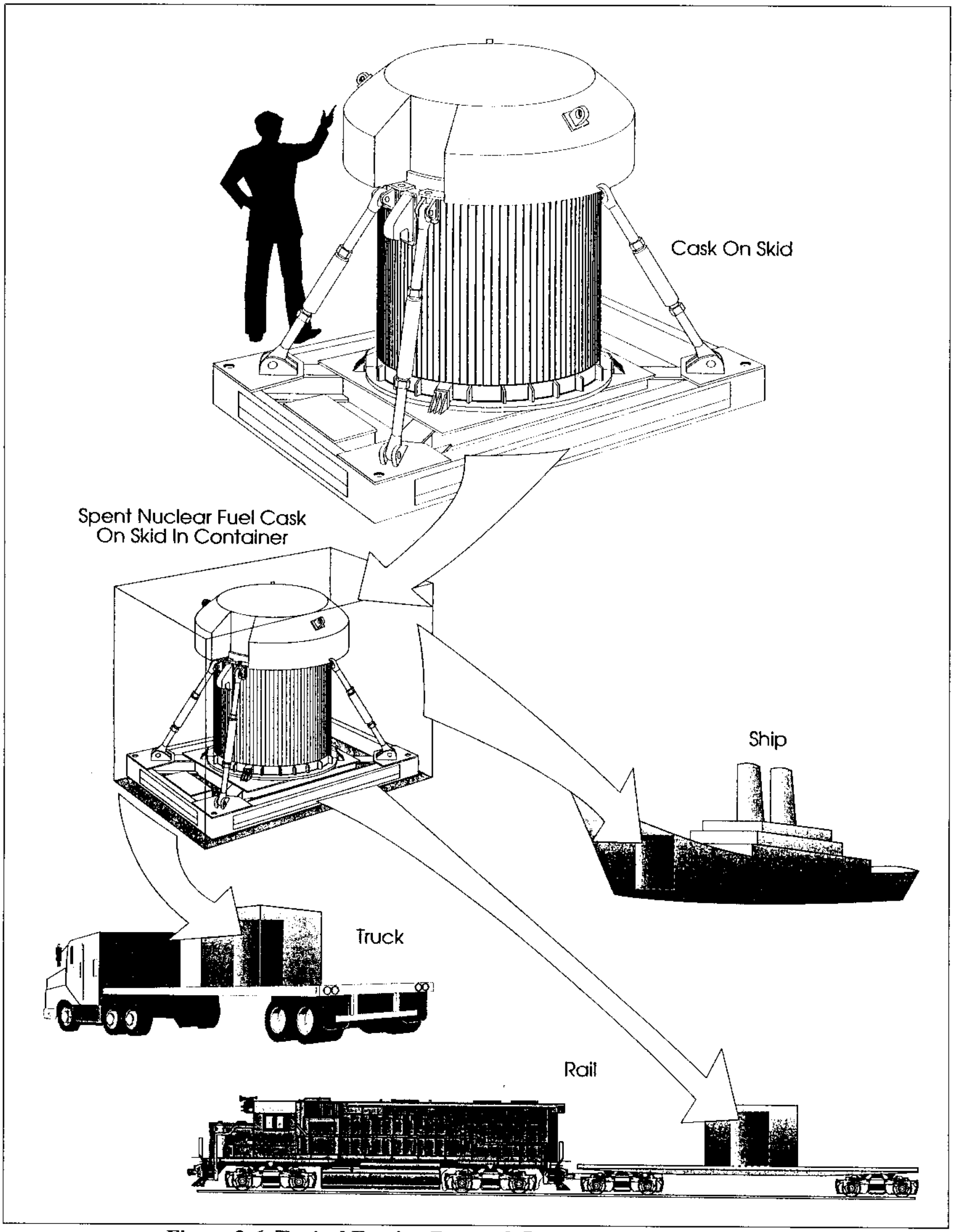


Figure 2-6 Typical Foreign Research Reactor Spent Nuclear Fuel Transportation Cask

Table 2-5 Representative Transportation Casks for Foreign Research Reactor Spent Nuclear Fuel

Transportation Cask	Country of Origin	Certificate of Compliance	Estimated Capacity Number of Elements
Marine Transport			
LHRL-120	Australia	USA/0389/B(U)F	114
GNS-11	Germany	USA/0381/B(U)F	21-33
TN-1	Germany	USA/0316/B(U)F	126
IU-04	France	USA/0100/B(U)F	36-40
TN-7 (TN-7/2)	Germany	USA/0130/B(U)F	60-64
NAC-LWT	United States	USA/9225/B(U)F	48-64
UNIFETCH	United Kingdom	GB/1113/B(M)F	24-40
GOSLAR	Germany	USA/0094/B(M)F	13
Ground Transport (Between Sites)			
NL1-10/24	United States	USA/9023/B( )F	120-160
IF-300	United States	USA/9001/B( )F	84-112
BMI-1	United States	USA/5957/B(U)F	24
GE-2000	United States	USA/9228/B(U)F	24
TN-8	Germany	USA/9015/B()F	36-48
NLI-1/2	United States	USA/9010/B( )F	48-64
NAC-LWT	United States	USA/9225/B(U)F	48-64

A full cask can carry from 13 to 120 spent nuclear fuel elements from foreign research reactors, depending on fuel element design, size, and cask capacity. The casks are certified as "Type B" under regulations. To receive this certification, a cask must successfully pass tests simulating severe accident conditions. The tests include being dropped onto an unyielding surface, dropped onto a steel post, subjected to extremely high temperatures of 802°C (1,475°F) for 30 minutes, and submersion in water for 8 hours.

As discussed in Section 2.7.2, "Type B" casks have been used for years to transport spent nuclear fuel elements within the United States and around the world (DOE, 1994d). To date, no spent nuclear fuel transportation cask has ever been punctured or released any of its radioactive contents, even in actual highway accidents (NRC, 1993).

The casks are designed to provide shielding from radiation. However, a low radiation field is present outside the cask — usually less than one mrem per hour at 1 m (3.3 ft) away from the cask.

Table 2-5 identifies typical transportation casks that could potentially be used for transporting foreign research reactor spent nuclear fuel from the foreign research reactor sites to the candidate United States ports of entry and to the potential management sites. A majority of these have already been used by DOE for transporting foreign research reactor spent nuclear fuel.

As explained in Section 2.6.4, the inability of certain management sites to accept foreign research reactor spent nuclear fuel at the beginning of the implementation period could necessitate temporary (as long as 10 years from the start of the policy) management of foreign research reactor spent nuclear fuel at an available site and the eventual transport of this foreign research reactor spent nuclear fuel to another site. At the time of the intersite transport, the foreign research reactor spent nuclear fuel elements would contain less radioactivity and less heat because of the decay process. The transportation, therefore, could be carried out in casks with larger capacity than those considered for marine transport. Such casks, currently licensed only for the transportation of commercial spent nuclear fuel in the United States, would need to be certified for foreign research reactor spent nuclear fuel. The size of these casks would allow the transport of up to 4 (truck-size casks) or up to 10 (rail-size casks) times as many elements in a single shipment as those considered for the marine transport casks. The bottom part of Table 2-5 identifies typical casks for intersite transportation. Description and design information is included in Appendix B (Section B.2).

2.6.3 Marine Transport and Ports

This section describes the potential activities related to foreign research reactor spent nuclear fuel marine port identification and marine transport activities.

2.6.3.1 Marine Port Identification

In this EIS, port screening and selection were performed to identify candidate ports of entry for the foreign research reactor spent nuclear fuel. The criteria used in this process were based on several sources, including:

- A DOE-sponsored workshop on port selection criteria for spent nuclear fuel held at the U.S. Merchant Marine Academy at Kings Point, NY, on November 15-16, 1993 (USMMA, 1994).
- Public input to the scoping meetings for this EIS, as summarized in the DOE Implementation Plan (DOE, 1994h).
- Factors identified in Section 3151 of the National Defense Authorization Act for Fiscal Year 1994.

These sources are described in more detail in Appendix D. After consulting the above-mentioned sources, a list of criteria for ports eligible to receive spent nuclear fuel was developed. These criteria are:

- Appropriate port experience - port terminal(s) and operators should routinely handle containerized dry cargoes that require the same type of handling as containerized spent nuclear fuel, or will have the capability to handle these cargo types during the proposed management policy period;
- Port transit - the port should be within reasonable distance from the open sea, with a good ship channel;
- Appropriate port facilities - the port should have adequate crane(s), piers, and depth of water alongside the pier;
- Ready intermodal access - the port should have ready access for intermodal transport; and
- Low human population - the human population of the ports and along transportation routes to potential management sites should be low to the extent feasible and maximum extent practicable.

These criteria, taken collectively, provided DOE and the Department of State with the basis for identifying and analyzing potential ports of entry. Additionally, port identification was expanded (i.e., the NWS Charleston was added to the Port of Charleston) as a result of public comment on the Draft EIS.

2.6.3.2 Marine Transport and Port Activities

2.6.3.2.1 Marine Transport

DOE and the Department of State estimate that approximately 721 cask loads of foreign research reactor spent nuclear fuel would be sent to the United States by ship over the 13-year acceptance period under the basic implementation of Management Alternative 1. The International Maritime Organization currently limits the typical commercial cargo ship (Class INF-2) to a maximum of 200 petrabecquerels of radioactivity (IMO, 1993), which equates to approximately 5.4 million Ci. A typical cask of foreign research reactor spent nuclear fuel is predicted to contain 1 million Ci (see Appendix C). Therefore, a shipment in a commercial cargo ship could contain several casks.

Two types of analysis were conducted to evaluate the impacts of the marine transport of foreign research reactor spent nuclear fuel: first, assuming there are no accidents (incident-free); second, assuming various accidents occur. The incident-free analyses were conducted for ships' crews and port workers, assuming ships carrying two and eight casks of foreign research reactor spent nuclear fuel. Accident analyses were conducted for accidents in port and for accidents in coastal waters and the open ocean. The number of shipments is a parameter of primary importance in the incident-free analysis as well as the accident analysis in port, coastal waters, and open ocean. As noted above, 721 shipments were considered for the basic implementation of Management Alternative 1. In implementing the proposed policy, DOE would attempt to minimize the number of shipments by maximizing the number of casks that would be carried in a single shipment. However, for the purpose of assessing the environmental impacts, a single-cask per shipment assumption is made for the purpose of conservatism. The number of shipments for the implementation alternatives discussed in Sections 2.2.2.1 through 2.2.2.7 and Management Alternative 3 discussed in Section 2.4 are roughly proportional to the amount of foreign research reactor spent nuclear fuel to be accepted in the United States under each alternative. The exact number of shipments assumed in the analysis is provided in Appendices C and D, Section C.4 and D.4, respectively. The results of both the incident-free and the accident analyses are presented in Chapter 4, with details in Appendices C and D.

There are four types of ships that could be used to transport foreign research reactor spent nuclear fuel casks. These are:

Container vessels: These are typically large ships specifically intended for the transport of containerized cargo. Some modern container ships can transport up to about 5,000 containers, although a more typical capacity is in the range of 800 to 1,000 containers. A principal advantage of container ships, because of standardization of containers, is that the vessel can be rapidly loaded or off-loaded at those ports equipped with container gantry cranes. Containers can be removed from, or placed on, the vessel at an average rate of about 45 containers per hour. At well-equipped container ports, two cranes are used to move containers.

Roll-on/roll-off ships: These ships are vehicle carriers used for the ocean transport of cars and trucks. The ships are loaded and unloaded using a ramp between the vessel and dock. Typically, the vessel carries its own ramp, which is deployed by an on-board crane, hydraulic cylinders, or chain drives. The ramp may extend from the stern of the vessel or from a hatch in the side of the vessel hull. At docks intended for roll-on/roll-off service, additional ramps may be deployed from the dock to expedite loading or unloading. This type of ship could carry foreign research reactor spent nuclear fuel casks secured on trailers.

General/cargo (breakbulk) ships: General cargo vessels are smaller ships that typically call on less well-developed or equipped ports. They have on-board jib or boom type cranes that can be used to load or unload the ship if dockside crane service is not available. As the name implies, these vessels are intended to accommodate a wide variety of cargoes that may have any reasonable configuration. Since the advent of the widespread use of containers, most of these ships are equipped with lock fixtures to secure containers during transport. If necessary, containers can be lifted on and off these ships by using four-legged slings between the corners of the container and hook of the crane.

Purpose-built ships: For the purposes of this EIS, the ships discussed here are specifically designed to transport spent nuclear fuel casks. These ships are not used for the transport of any other cargo, and they operate as chartered vessels. Casks are loaded directly into the holds of the ship because the cargo compartments contain the hardware needed to mate with the tiedown fixtures of the cask. If the ship has no crane, dockside cranes are used for loading and unloading. The cargo compartments are typically intended to handle only one cask type, however, other casks may be used with minor modifications. For the relatively efficient transport of spent nuclear fuel, the casks are large. These type vessels are intended for the transport of commercial power nuclear reactor fuel, and they generally operate between nuclear installations (power plants and spent nuclear fuel end-use facilities) having dedicated docks. Commercial docks are not normally used, but could be. These vessels have double bottoms and hulls and collision damage-resisting structures within the hull. The vessel crew is trained in the handling of the cargo and in emergency response like most other commercial vessels.

The potential exists that spent nuclear fuel would be accepted from all 41 countries that have expressed interest in this program. Ships carrying the foreign research reactor spent nuclear fuel would follow normal shipping routes from a convenient port in or near the country of origin, and would go to a U.S. port that is consistent with the port identification, evaluation, and selection process as described in Appendix D.

Regularly scheduled commercial service cargo ships could be used to ship foreign research reactor spent nuclear fuel. Some, if not most, of the regularly scheduled commercial ships might initially call at a port other than the port of destination of the foreign research reactor spent nuclear fuel, and may make additional stops. Therefore, marine transport may involve entry into and departure from intermediate ports and shipping in coastal waters. Typically, ships spend 1 day in each port of call and 1 or 2 days passing between ports.

Risks to the ships carrying foreign research reactor spent nuclear fuel and to the spent nuclear fuel itself can arise from natural sources, such as storms at sea, and from other events, such as collisions with other ships and marine obstacles, as well as from fires. Modern technology and good communications help minimize these risks by keeping ships informed of severe weather and other shipping and marine obstacles. Risk to the cargo is further reduced through proper stowing and securing, and through daily cargo inspections while at sea to ensure that the cargo remains secured.

Regardless of the technology and practices mentioned above, accidents involving ships carrying foreign research reactor spent nuclear fuel would be possible. Consequences of accidents at sea have been evaluated and are discussed in Chapter 4, and described in more detail in Appendix C.

The presence of a cask containing foreign research reactor spent nuclear fuel onboard could result in a radiation dose to some of the ship's crew due to radiation that emanates from the cask. Most of the ship's crew would be relatively far away from the cargo (and the cask), and therefore, would receive essentially no radiation dose. However, the daily inspection of the cargo would bring an inspector in close proximity to the cask containing the foreign research reactor spent nuclear fuel for a short period of time. Effects on inspectors and other incident-free impacts were evaluated, and are described in Chapter 4 and detailed in Appendix C.

Both commercial and military ports were evaluated for potential use as ports of entry for the foreign research reactor spent nuclear fuel. DOE determined that the security provisions specified by 10 CFR 73, which are required for all spent fuel shipments, could be implemented at either commercial or military ports. Any additional security that might be available at a military port would not be required for foreign research reactor spent nuclear fuel shipments.

2.6.3.2.2 Port Activities

Entry into a port by commercial vessels is accomplished under the control of the port authority. The port authority is responsible for the area from the sea buoy to the dock, except where the approaches are long, such as the case with the Chesapeake Bay. Normally, each ship is required to have a pilot familiar with local conditions to direct it while underway in the harbor or channel and during both entry and exit approaches. The pilot's job is to ensure that the ship follows the marked channel and arrives safely at its dock or other assigned location. In the event of bad weather or low visibility, radar and other instruments are available on all ships that would be considered for carrying foreign research reactor spent nuclear fuel.

In most cases, the ship moves directly to its assigned dock. However, if the assigned dock is still occupied or is not immediately available for other reasons, the ship may anchor in the harbor or its approaches for a period of time prior to docking. Most ocean-going vessels are not highly maneuverable in confined spaces, so docking is normally accomplished with the help of one or more tugs.

At the ship's first port of call in the United States, the U.S. Coast Guard and other authorities would inspect the ship, its cargo, and documentation. In the case of radioactive cargoes, the NRC may inspect the container with the radioactive material and its documentation. State and local officials could also perform inspections of documents and cargo. At ports of call after the initial port(s) of entry, additional inspections may be performed by Federal, State, or local officials.

Except for roll-on/roll-off ships, all cargo ships that would potentially carry foreign research reactor spent nuclear fuel are unloaded with cranes. Unloading of a foreign research reactor spent nuclear fuel cask, whether in a container or not, involves connecting a lifting fixture to the container or to the cask pallet, lifting the container or cask, and placing it dockside, either on an intermediate vehicle or directly on the primary mode of land transportation. Typically, container ships can be unloaded at the same rate as they are loaded (approximately 45 containers per hr), while unloading a cask on a handling platform in a breakbulk ship would require more time.

All shipments of foreign research reactor spent nuclear fuel would be anticipated well in advance, so the container housing the foreign research reactor spent nuclear fuel cask would normally be loaded immediately on the ground transportation to be used to carry it out of the port. Should there be a delay, the container may be temporarily stored at the port for up to 24 hours. Port security at any of the ports selected for analysis is adequate to protect the container in the event of this unexpected delay.

In spite of all of the precautions taken, accidents in the port would be possible. In fact, most ship accidents occur in or around ports. DOE has had no radioactivity released in the past due to port accidents; however, a range of accidents, both at the dock and in the port or its approaches, has been evaluated. See Chapter 4 for a discussion of the results of these analyses.

All ports considered for receiving foreign research reactor spent nuclear fuel would have emergency plans for responding to an accident in the port.

2.6.4 Ground Transport Route Options and Route Identification Process

2.6.4.1 Ground Transport Route Options

Route options for the potential ground transportation of foreign research reactor spent nuclear fuel depend on the marine ports considered, the management sites and the various ways that foreign research reactor spent nuclear fuel would be distributed among the potential management sites according to the alternatives considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c). Accordingly, routes and the amount of fuel to be shipped would be established based on one of the following spent nuclear fuel distributions:

- an even distribution of foreign research reactor spent nuclear fuel between the Savannah River Site and the Idaho National Engineering Laboratory under the Decentralization and the 1992/1993 Planning Basis alternatives;
- a distribution that sends TRIGA spent nuclear fuel to the Idaho National Engineering Laboratory and aluminum-based spent nuclear fuel to the Savannah River Site under the Regionalization by Fuel Type alternative;
- a distribution that sends the spent nuclear fuel entering the United States from the Eastern ports to the Savannah River Site or the Oak Ridge Reservation and the spent nuclear fuel entering the United States from the Western ports to the Idaho National Engineering Laboratory, the Nevada Test Site, or the Hanford Site under the Regionalization by Geography alternative; or
- a distribution that sends all foreign research reactor spent nuclear fuel to one of the five potential management sites under the Centralization alternative.

For the purposes of this EIS, the distribution of foreign research reactor spent nuclear fuel between sites under Regionalization by Geography and by Fuel Type has been analyzed in detail. The more detailed planning performed in preparation for the analyses of the various alternatives considered in this EIS did not reveal any physical situation in which an even distribution of the spent nuclear fuel between two sites was advantageous. Furthermore, the impacts of activities associated with the even distribution at either site would be bounded by and equal to roughly 50 percent of the impacts of the centralization of all foreign research reactor spent nuclear fuel management activities at that site. As a result, this alternative is not analyzed in detail in this EIS.

An additional factor which would affect the route options for ground transportation is the inability of certain potential spent nuclear fuel management sites to implement the foreign research reactor spent nuclear fuel management policy immediately. Of the five sites, only two (the Savannah River Site and the Idaho National Engineering Laboratory) would be immediately available in late 1995. The other three could become available at a later date when appropriate facilities for accepting and managing foreign research reactor spent nuclear fuel become available. This constraint affects the ground transportation route options in the case that a site, other than the Savannah River Site or the Idaho National Engineering Laboratory, is considered and for DOE's spent nuclear fuel management under either the Regionalization by Geography or the Centralization alternative. If the Nevada Test Site, the Oak Ridge Reservation, or the Hanford Site is one of the management sites, the foreign research reactor spent nuclear fuel would have to be shipped first to one of the available management sites (the Savannah River Site and/or the Idaho National Engineering Laboratory) and later, when appropriate facilities are completed, to either the Nevada Test Site, the Oak Ridge Reservation, or the Hanford Site.

Certain assumptions are required in order to simply and consistently describe the manner in which foreign research reactor spent nuclear fuel would be transported to the management sites. The shipments, which were identified earlier in Tables 2-1 and 2-2, were divided into east coast and west coast shipments, depending on the country of origin. Spent nuclear fuel shipments from Europe, Africa, the Middle East, and parts of Central and South America were designated as east coast shipments. All others were designated as west coast shipments. Shipments from Canada were assumed to enter the United States from either an eastern or western point of entry, depending on the point of origin in Canada. Under these assumptions, for the basic implementation of Management Alternative 1, the Eastern points of entry would receive 651 cask shipments (535 from ports, 116 from Canada) and the Western ports of entry would receive 186 cask shipments (all from ports).

No intersite shipments would be necessary under the Programmatic SNF&INEL Final EIS alternatives (DOE, 1995c) that use the Savannah River Site and/or the Idaho National Engineering Laboratory for managing the foreign research reactor spent nuclear fuel. The estimated number of shipments for the basic implementation of Management Alternative 1 in these cases would be as follows:

- Decentralization, 1992/1993 Planning Basis, or Regionalization by Geography to the Savannah River Site and the Idaho National Engineering Laboratory - the Savannah River Site would receive 651 casks from the east coast and the Idaho National Engineering Laboratory would receive 186 casks from the west coast.

- Regionalization by Fuel Type - the Savannah River Site would receive 675 casks of aluminum-based fuel; 544 from the east coast and 131 from the west coast. The Idaho National Engineering Laboratory would receive 162 casks of TRIGA-type fuel; 107 from the east and 55 from the west.

- Centralization to the Idaho National Engineering Laboratory or Centralization to the Savannah River Site - the site would receive 837 casks; 651 from the east coast and 186 from the west coast.

A two-phased program would be required if a site other than the Idaho National Engineering Laboratory or the Savannah River Site is considered as a central or regional site. Phase 1 is defined as the period from the beginning of the policy (late 1995) until the Phase 2 site (the Hanford Site, the Nevada Test Site and/or the Oak Ridge Reservation) would be ready to receive fuel, which is estimated to be 10 years for new construction; less time would be required for refurbishment of an existing facility. During Phase 1, DOE would manage the fuel at the Savannah River Site and/or the Idaho National Engineering Laboratory. During Phase 2, DOE would ship any fuel that is being managed during Phase 1 at a non-Phase 2 site to a Phase 2 site, and manage the fuel at that site until a repository becomes available. The phases are defined to help describe the implementation of the foreign research reactor spent nuclear fuel management policy and to analyze the transportation impacts of the implementation of the policy.

If the Hanford Site, the Nevada Test Site, and/or the Oak Ridge Reservation were selected under the Programmatic SNF&INEL EIS, DOE and the Department of State would select from the following four strategies for managing fuel at the Savannah River Site and/or the Idaho National Engineering Laboratory during Phase 1. DOE could: (1) divide the fuel by geography, (2) divide the fuel by type (aluminum-based and TRIGA), (3) ship all fuel to the Savannah River Site, or (4) ship all fuel to the Idaho National

Engineering Laboratory. Therefore, in Phase 2, the Hanford Site and the Nevada Test Site could receive all foreign research reactor spent nuclear fuel, or TRIGA or Western spent nuclear fuel managed at the Idaho National Engineering Laboratory during Phase 1. Similarly, the Oak Ridge Reservation could eventually receive all foreign research reactor spent nuclear fuel or the aluminum-based or Eastern spent nuclear fuel managed at the Savannah River Site during Phase 1.

An assumption on the rate at which spent nuclear fuel arrives is necessary to estimate the number of shipments that would arrive during Phases 1 and 2. The demand to ship fuel by the foreign research reactor operators would be highest at the beginning and the end of the proposed policy period. However, the limited availability of casks would compel DOE to receive fuel at a steady rate. DOE could control the rate at which fuel is delivered by managing the contracts with shippers. Therefore, for the purposes of this analysis, it would be reasonable to assume that the 837 casks would arrive at a uniform rate during a 13-year period. Based on this rate of about 65 casks per year, it is estimated that 644 casks would be received during Phase 1 (approximately 10 years), and 193 casks would be received during Phase 2.

The projected number of shipments for the two-phased regionalization and centralization approaches are shown in Tables 2-6 and 2-7. The projections are based on the types and locations of spent nuclear fuel described in Appendix B, and the strategies and arrival rate assumptions described above. Each projection is described in more detail and shown on a map in Appendix E.

As noted earlier, the impact analysis from transportation depends on the location of entry (Eastern or Western ports) and number of shipments that would reach the United States. The discussion above pertains to the basic implementation of Management Alternative 1. In considering the implementation alternatives discussed in Sections 2.2.2.1 through 2.2.2.7 and Management Alternative 3, discussed in Section 2.4, both the number of shipments and locations of entry would vary with each alternative. The detailed distribution and number of shipments assumed to set up the ground transportation routes for each alternative are provided in Appendix E, Section E.8.

2.6.4.2 Route Analysis

Foreign research reactor spent nuclear fuel shipments would have to comply with both NRC and Department of Transportation regulatory requirements. The highway routing of spent nuclear fuel is systematically determined in accordance with Department of Transportation regulations [49 CFR 171-179 and 49 CFR 397].

The Department of Transportation routing regulations require that these shipments be transported over a preferred highway network including:

- Interstate highways;
- An interstate system bypass or beltway around a city; or
- State-designated preferred routes.

The selection of the preferred highway routes are consistent with the U.S. Department of Transportation's published guidelines (DOT, 1992).

In addition to defining routes, 49 CFR Part 397 contains the driver safety requirements for highway carriers of packages of radioactive material exceeding a quantity of material known as a "highway route-controlled quantity." All spent nuclear fuel shipments would be expected to exceed this quantity.

Table 2-6 Shipment Summary for Regionalization by Geography Alternatives

Spent Nuclear Fuel Site Option Western/Eastern	Phase 1 Approach	Phase 1 Port-to-Site Shipments	Site-to-Site Shipments[a]	Phase 2 or Port-to-Final Site Shipments	Total Number of Shipments[a]
INEL/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51	East to ORR: 150 West to INEL: 43	963/888
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52	East to ORR: 150 West to INEL: 43	967/889
	All to INEL	644	None	East to ORR: 150 West to INEL: 43	837
NTS/SRS	Geographic	East to SRS: 501 West to INEL: 143	INEL to NTS: 36/15	East to SRS: 150 West to NTS: 43	873/852
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	INEL to NTS: 31/13	East to SRS: 150 West to NTS: 43	868/850
	All to SRS	644	None	East to SRS: 150 West to NTS: 43	837
NTS/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51 INEL to NTS: 36/15	East to ORR:150 West to NTS: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52 INEL to NTS: 31/13	East to ORR: 150 West to NTS: 43	998/902
	All to SRS	644	SRS to ORR: 161/65	East to ORR: 150 West to NTS: 43	998/902
	All to INEL	644	INEL to NTS: 161/65	East to ORR: 150 West to NTS: 43	998/902
HS/SRS	Geographic	East to SRS: 501 West to INEL: 143	INEL to HS: 36/15	East to SRS: 150 West to HS: 43	873/852
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	INEL to HS: 31/13	East to SRS: 150 West to HS: 43	868/850
	All to SRS	644	None	East to SRS: 150 West to HS: 43	837
HS/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51 INEL to HS: 36/15	East to ORR: 150 West to HS: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52 INEL to HS: 31/13	East to ORR: 150 West to HS: 43	998/902
	All to SRS	644	SRS to ORR: 161/65	East to ORR: 150 West to HS: 43	998/902
	All to INEL	644	INEL to HS: 161/65	East to ORR: 150 West to HS: 43	998/902

SRS = Savannah River Site, INEL = Idaho National Engineering Laboratory, HS = Hanford Site, ORR = Oak Ridge Reservation, NTS = Nevada Test Site

[a] *Truck/rail shipments, assuming that the truck casks used for interstate shipments are capable of carrying 4 times as much fuel, and rail casks 10 times as much fuel as the shipping cask received from the foreign research reactor.*

Rail routing is not covered by specific Department of Transportation and NRC regulations. Therefore, carriers would generally select the most direct route, which would serve to reduce travel time and radiation exposure consistent with track class and other rail service requirements.

NRC regulations concerning physical security and notification are set forth in 10 CFR 71 and 10 CFR 73, respectively. Carriers are required to submit proposed routes for spent nuclear fuel shipments to NRC for approval, and NRC publishes a public information circular that lists routes that have been evaluated and approved for specific spent nuclear fuel shipments (NRC, 1993).

Table 2-7 Shipment Summary for Centralization Alternatives

Spent Nuclear Fuel Site Option	Phase 1 Approach	Phase 1 Port-to-Site Shipments	Site-to-Site Shipments[a]	Phase 2 or Port-to-Final Site Shipments	Total Number of Shipments[a]
SRS	N/A - Single phase			837	837
INEL	N/A - Single phase			837	837
HS	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902
ORR	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902
NTS	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902

SRS = Savannah River Site, INEL = Idaho National Engineering Laboratory, HS = Hanford Site, ORR = Oak Ridge Reservation, NTS = Nevada Test Site

[a] *Truck/rail shipments assuming that the truck casks used for intersite shipments are capable of carrying 4 times as much fuel and rail casks 10 times as much fuel as the shipping cask received from the foreign research reactor due to consolidation.*

The HIGHWAY and INTERLINE computer codes are used to assist in route selection and estimations of exposed population (DOE, 1995c). The collective population risk, maximally exposed individual (MEI) risk, accident risk, accident consequence, and nonradiological risk assessments are performed using the RADTRAN and RISKIND computer codes established for shipment by both railroad and highway. Additional details of the treatment and analysis methodology used in the ground transportation assessment are given in Appendix E.

2.6.5 Activities and Alternatives at the Foreign Research Reactor Spent Nuclear Fuel Management Sites

The potential sites for receipt and management of foreign research reactor spent nuclear fuel are the same as those considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c), namely: the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site.

Since foreign research reactor spent nuclear fuel is part of the overall DOE spent nuclear fuel management program, the potential site-specific options are consistent with the site management alternatives considered in the Programmatic SNF&INEL Final EIS. The alternatives are: Decentralization and 1992/1993 Planning Basis (even distribution of foreign research reactor spent nuclear fuel between the Idaho National Engineering Laboratory and the Savannah River sites), Regionalization (distribution by fuel type and geography), and Centralization (all foreign research reactor spent nuclear fuel at the potential site).

As discussed earlier, the site-specific foreign research reactor spent nuclear fuel management options also depend on the availability of the management sites to implement the policy immediately. Of the five sites, only the Savannah River Site and the Idaho National Engineering Laboratory will be available in late 1995. The other three could become available at a later date when construction or refurbishment of appropriate facilities is completed. This constraint has resulted in the two-phased approach considered in some cases (see discussion in Section 2.6.4.1). For the purpose of the site impact analysis, the implementation of the policy was divided into two functional periods — the period during which receipt and management of foreign research reactor spent nuclear fuel is accomplished by using existing facilities (Phase 1), and the period during which new or refurbished facilities are used (Phase 2). For the environmental impact analysis, the first is characterized by operational activities only, while the second involves impacts from construction and operation activities.

Section 2.6.5.1 provides an overview of the storage technologies and descriptions of the storage facilities considered under the implementation alternatives of Management Alternatives 1 and 3. Section 2.6.5.2 provides a description of chemical separation, which is considered as an implementation alternative to storage.

The site-specific options selected for impact analysis are described separately in the sections devoted for each site (Sections 2.6.5.3.1 through 2.6.5.3.5).

2.6.5.1 Storage Technologies

The purpose of a spent nuclear fuel management facility is to provide an environment for the storage of spent nuclear fuel that protects the public, onsite workers, and the environment. The principal hazard presented by spent nuclear fuel is its inventory of radioactive elements that are the products of the reactions in a nuclear reactor. In addition, the fissionable uranium and plutonium remaining in the spent nuclear fuel has the potential of sustaining a fission chain reaction, which would generate additional radiation and fission products.

The management facility is designed to prevent the stored spent nuclear fuel from achieving a fission reaction (termed "criticality") and to isolate the radioactive materials within the spent nuclear fuel from the public and workers. Criticality is prevented by such methods as:

- maintaining a minimum separation distance between adjacent spent nuclear fuel elements;
- limiting the concentration of fissionable materials in each spent nuclear fuel storage container;
- installing neutron-absorbing materials between spent nuclear fuel elements; and
- controlling the presence and/or concentration of other materials that would enhance the ability of the stored spent nuclear fuel to become critical.

Protection of the public and workers from the radioactive materials within each spent nuclear fuel element is achieved by:

- enclosing or encapsulating the spent nuclear fuel so that any accidental release of radioactive material is retained;
- maintaining a benign chemical and thermal environment around the spent nuclear fuel so that its structural integrity is preserved;
- providing adequate shielding of the radiation emanating from the spent nuclear fuel so that dose rates outside the facility are lowered; and
- utilizing security barriers to isolate spent nuclear fuel from workers and public.

The technology for safely storing spent nuclear fuel (as defined by the above criteria) has been in use, in one form or another, for over 40 years in the nuclear industry. Spent nuclear fuel storage is generally characterized as either wet or dry, denoting whether the spent nuclear fuel elements reside in a water-filled pool or a dry atmosphere. Details of the concepts are provided in Appendix F, Section F.1.

The wet pool type of spent nuclear fuel storage is used at almost every water-cooled nuclear reactor in the world. There are currently more than 600 operating water-cooled power and research nuclear reactors, each with an individual storage pool. The pool design uses common materials (water and concrete) for spent nuclear fuel shielding, heat removal, and the confinement of any radioactive material that might be released from the spent nuclear fuel. An additional benefit is the ability to visually inspect spent nuclear fuel, since the water purity and clarity are maintained at a high level. Spacing, fissionable material limits, and in some cases, the use of neutron-absorbing material prevent criticality in a wet storage environment. The pool is enclosed in a suitably qualified structure or building. Construction of a wet storage facility involves excavating earth, backfilling, pouring concrete, setting piping, erecting a building around the pool, and installing piping, electrical systems, and heating, ventilating, and air conditioning systems. In many ways, a spent nuclear fuel storage pool is like a swimming pool, except its depth is greater and its concrete walls and floors are much thicker to provide for structural integrity. Wet storage facility designs include sophisticated methods of leak detection. To negate corrosion, the pool water purity and quality are carefully maintained and controlled.

Dry storage technology involves the encapsulation of spent nuclear fuel in a steel cylinder that may be placed in a concrete or massive steel cask or structure. The spent nuclear fuel is stored in racks within the cylinder or suspended from plates placed at variable distances in the cylinder, in either air, or inert atmosphere. Foreign research reactor spent nuclear fuel elements with suspect cladding integrity would be placed in sealed cans before they are placed in the cylinder (canning). Casks or structure materials, usually some form of concrete, steel, iron, or lead provide shielding and heat removal. Spacing, fissile material limits, and neutron absorbing materials are used to prevent criticality. Different forms of dry fuel storage have been used for over 40 years in the nuclear industry. Several nuclear power plants in the United States have licensed, built, and operated dry storage facilities during the last 7 years. NRC has reviewed and approved several manufacturers' designs for dry fuel storage of commercial spent nuclear fuel. Canada has been storing spent commercial nuclear power plant fuel in dry storage casks since 1975. Australia has been successfully storing its research reactor spent nuclear fuel since 1963 (Silver, 1993) in dry environment, and Japan has had 12 years of experience with dry storage of research reactor spent nuclear fuel (Shirai et al., 1991). The Savannah River Site has an ongoing developmental program on dry storage technology which would be used to implement this worldwide experience in the United States, and to finalize design parameters for a foreign research reactor spent nuclear fuel dry storage facility.

Dry storage methods are not as efficient in removing heat from the spent nuclear fuel as wet storage pools. Thus, as explained in Appendix F, this EIS assumes that high decay heat foreign research reactor spent nuclear fuel would initially be placed in wet storage. This would allow sufficient time for the spent nuclear fuel radioactive decay heat to decrease and not be a deciding factor in sizing a dry storage facility.

Dry storage facility construction involves the preparation and pouring of concrete foundations upon which the concrete or metal cask or building is then erected. Metal casks would be built away from the DOE site in a factory, since they involve thick metal fabrication techniques not used at DOE facilities. Concrete casks or buildings are constructed at the site using the same general principles (e.g., forms, rebar) as in nonnuclear concrete construction. Qualified concrete foundation pads are also poured for support bases of the casks.

Whether wet or dry storage were used, the facility would be designed to withstand natural phenomena such as earthquakes, floods, tornadoes, hurricanes, high and low temperatures, and wind generated missiles (branches, poles, etc.). The design would also include provisions to preclude sabotage or terrorist acts. Security requirements for a dry storage facility after the spent nuclear fuel has decayed to low levels of radioactivity and is no longer self-protecting would be met by the establishment of a Perimeter Intrusion Detection and Alarm System zone, which is the standard procedure for DOE. Each design has specific provisions for periodic inspection or surveillance, and must meet the highest quality standards associated with all safety requirements specified for nuclear facilities.

The current alternative types of storage technology are discussed and evaluated in detail in Appendix F. The basic categories are: wet (pool), dry concrete vault or building, dry concrete horizontal cask/module, dry concrete vertical cask/silo, dry metal vertical cask, hot cells, multi-purpose casks, and dry inground vertical holes. There are significant differences between these technologies in terms of construction, operations and maintenance costs and various design details. However, these differences do not result in any important variations in environmental impacts and consequences. With the exception of multi-purpose casks (which are still under development), all of these technologies have proven records of successful safe operation while storing spent nuclear fuel. Appendix F provides detailed descriptions concerning generic dry and wet storage facilities for foreign research reactor spent nuclear fuel. Brief descriptions of both wet and dry storage facilities are provided in the following sections.

2.6.5.1.1 Description of Dry Storage Facilities

Spent Nuclear Fuel Storage Using a Modular Dry Vault:

An aboveground dry vault is a self-contained concrete structure that would allow for dry spent nuclear fuel handling and storage. This design represents an integrated spent nuclear fuel storage approach and would consist of four major components: a receiving/loading/inspection area, spent nuclear fuel storage canisters, a shielded canister handling machine, and a modular array for storing the spent nuclear fuel storage canisters. Figure 2-7 displays an illustration of a typical modular dry vault storage facility. The receiving area would use a wet pool for unloading the casks and for short-term (1 to 3 years) storage of foreign research reactor spent nuclear fuel elements with a heat load exceeding 40 Watts per element. The vault would consist of several modular units, and each unit could provide storage for hundreds of spent nuclear fuel assemblies. The vault itself would contain a charge/discharge bay with a spent nuclear fuel handling machine above a floor containing steel tubes that house the (removable) spent nuclear fuel canisters. The bay would be shielded from the stored spent nuclear fuel by the thick concrete floor and shield plugs inserted into the top of the steel storage tubes. The steel tubes would serve as secondary containment for the foreign research reactor spent nuclear fuel and would descend into an open storage

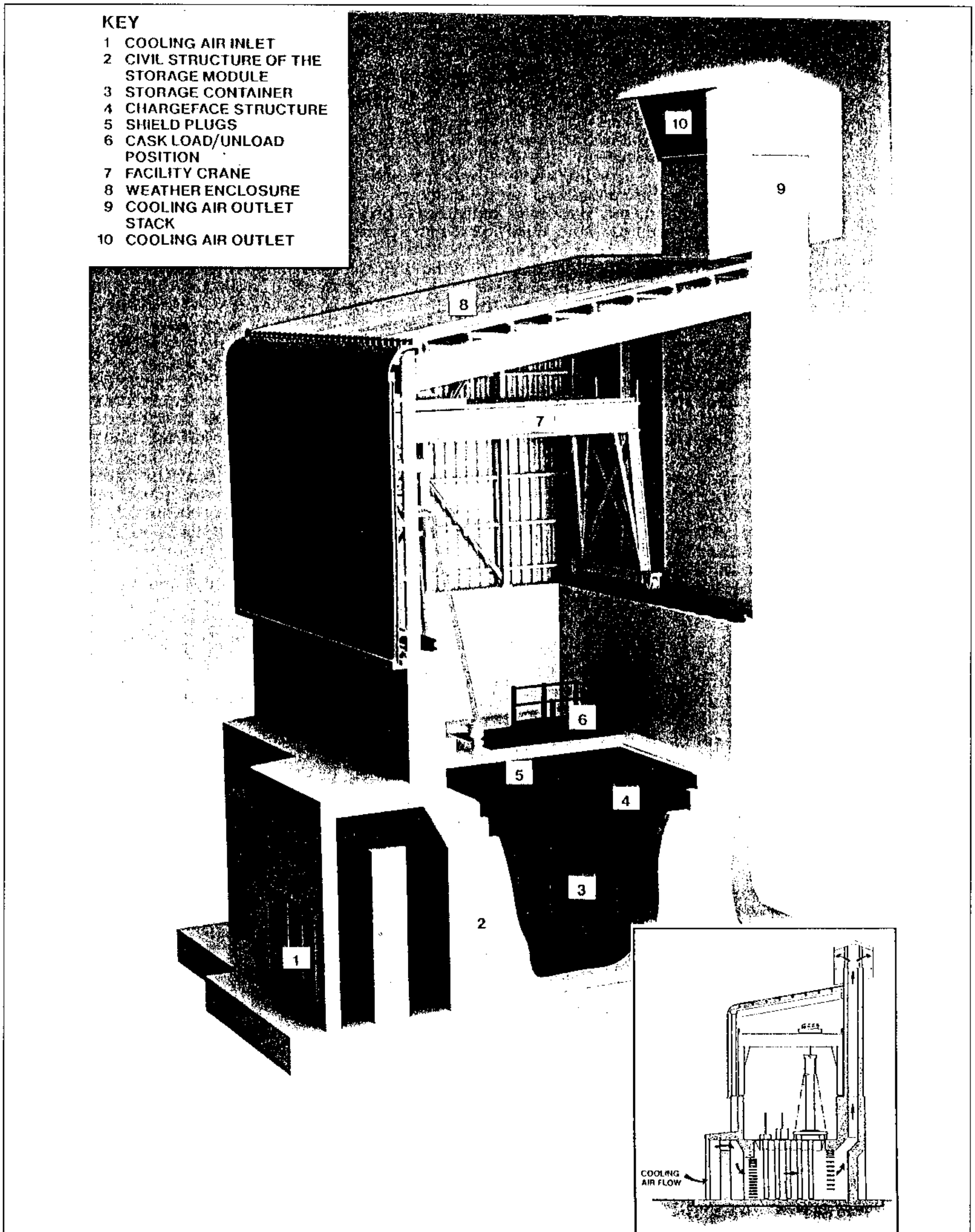


Figure 2-7 Illustration of a Typical Modular Dry Vault Storage Facility

Table 2-8 Summary of Modular Dry Vault Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel[a]

Construction Phase:	
Disturbed Land Area	3.7 ha (9 acres)
Facility: size (area) concrete steel	 5,000 m^2 (54,000 ft^2) 21,800 m^3 (28,500 yd^3) 5,200 metric tons (5,750 tons)
Soil Moved	11,000 m^3 (14,400 yd^3)
Equipment Fuel	835,000 l (221,000 gal)
Construction Debris/Waste	1,800 m^3 (2,400 yd^3)
Work Force	190/yr (average), 234/yr (peak)
Duration (years)	4 years for construction, 1.5 years for design
Capital Cost	$370 million[b]
Operation Phase:	
Electricity	800 - 1,000 MW-hr/yr (staging facility)
Water	2.1 million l/yr (550,000 gal/year) during receipt 0.9 million l/yr (238,000 gal/yr) thereafter
Wastestreams Solid Low-Level Waste Waste Water	 22 m^3/yr (780 ft^3/yr) during receipt 1 m^3/yr (35 ft^3/yr) thereafter 1.59 million l/yr (420,000 gal/yr) during receipt 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30 during receipt 8 thereafter
Annual Operating Cost	$15.6 million during handling, $0.6 million during storage[b]

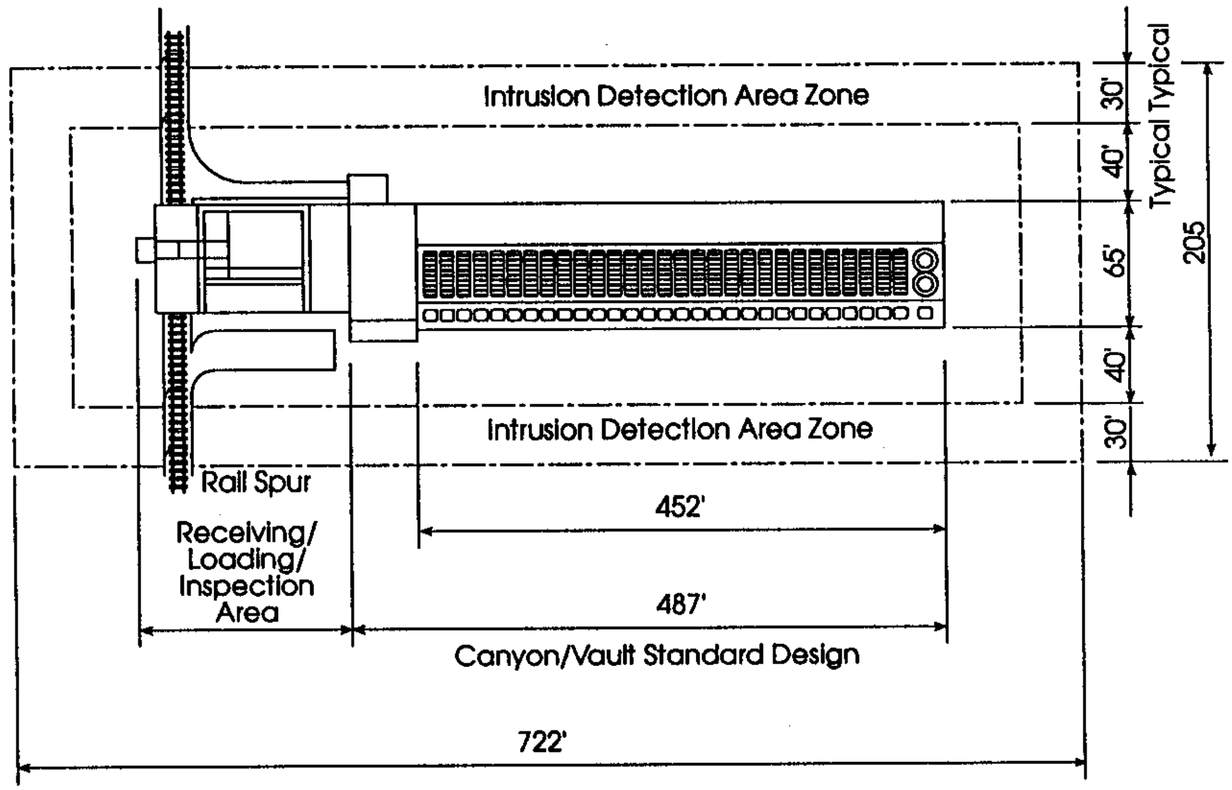
[a] *Staging facility parameters are based upon the regionalized, small wet pool (Dahlke, et al., 1994)*

[b] *Cost estimates are in 1993 dollars (EG&G, 1993)*

area. Large, labyrinth air supply ducts and discharge chimneys would permit natural convection cooling of the steel spent nuclear fuel storage tubes, while the perimeter concrete walls would provide for shielding. The design would allow for expansion by adding additional units of arrays to the end of the vault or by construction of another vault. The vault facility would also include a receiving and loading bay that would allow handling of shielded transportation casks and unloading of the foreign research reactor spent nuclear fuel into the short-term wet storage pool. The receiving bay provides for spent nuclear fuel inspection, canning as required and could be used for spent nuclear fuel characterization with additional equipment and modifications. Although it is not expected that the physical condition of the foreign research reactor spent nuclear fuel elements would require extensive canning, the capability of canning the entire foreign research reactor spent nuclear fuel inventory would be provided by the design. Table 2-8 summarizes modular dry vault storage parameters for foreign research reactor spent nuclear fuel storage.

In operation, the transportation cask would be lifted by a crane and placed in the unloading area of the small wet pool. The fuel elements would be removed underwater, examined, and if the heat generation rate is below 40 Watts per element, the spent nuclear fuel would be placed within the transfer canister. The transfer canister would be subsequently drained, dried, and seal-welded. The handling machine then would place the spent nuclear fuel inside of the spent nuclear fuel storage canister, and would transport the loaded canister to the storage tubes. The handling machine would include radiation shielding. Heat dissipation would be accomplished by natural convection from the surfaces of the handling machine and canister. Decay heat would be dissipated by natural convection: air would enter through inlet ducts at the bottom of the vault module, pass around the outside of the steel storage tubes containing the spent nuclear

Note:
All dimensions shown are approximate and may only be used for a conceptual "footprint" estimate.



10-40 W/Element Scenario
25,000 FRR spent fuel element modular array (Nominal) Storage req. (29 modules with 44 tubes per module - 20 spent fuel elements per tube)
• Passive cooling • Monitored storage canisters • Retrievable for periodic inspection • Self-contained facility for staging and storage of spent fuel elements

Figure 2-8 Layout of a Modular Dry Vault Storage Facility for Foreign Research Reactor Spent Nuclear Fuel (10 Watt to 40 Watt Element Basis)

fuel canisters, and exit through outlet ducts at the top of the module. Therefore, the vault would be a complete, integrated facility with all of the required capabilities for foreign research reactor spent nuclear fuel handling and storage.

The vault facility would store spent nuclear fuel in canisters that are approximately 40.6 cm (16 in) in diameter by 4.6 m (15 ft) long. As currently envisioned, foreign research reactor spent nuclear fuel would be stored within the canister in 5 levels with 4 elements per level, for a total of 20 spent nuclear fuel elements per canister (MTR-type design). The vault design would allow for 36 to 44 canisters per array unit, depending upon the decay heat of the spent nuclear fuel and a cladding temperature limit nominally 175°C (347°F) for aluminum-cladding with an air inlet temperature of 49°C (120.2°F). Thus, the number of vault units/arrays required for the storage of elements having a decay heat between 10 Watts and 40 Watts per element would be 27.

Most of the foreign research reactor spent nuclear fuel is expected to have decay heats between 10 Watts and 40 Watts per element. For "cold" fuel (less than 10 Watts per element), potentially more than 44 spent nuclear fuel canisters could be placed per vault unit. However, this would require a customized design, which could unnecessarily increase costs and implementation time. Figure 2-8 displays the layout of the modular dry vault storage facility (10 Watt to 40 Watt element basis).

Criticality concerns would be addressed primarily by the tube spacing in the vault. Borated concrete could also be used. For foreign research reactor spent nuclear fuel, criticality would not be expected to be a significant concern because a considerable fraction of the fissile uranium would have been consumed, and neutron-absorbing fission products would be present.

This vault design, without a pool, has been licensed by NRC for the Fort St. Vrain nuclear power plant site. It represents a complete, stand-alone facility that could be dedicated to foreign research reactor spent nuclear fuel without requiring the utilization of any other facilities at the host site. Cask handling, spent nuclear fuel transfer to a canister, and spent nuclear fuel storage could be accomplished within the facility. Additional facilities or modifications to the inspection area, including a pool, would be required for foreign research reactor spent nuclear fuel characterization.

The cost to construct a modular dry vault storage facility with a staging area sufficient to unload, characterize, can, temporarily store in a small pool, and transfer the spent nuclear fuel to the vault storage area is estimated to be $370 million. The annual operating cost for this facility is estimated to be $15.6 million during the period of handling and transfers of the spent nuclear fuel and $0.6 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993) with the addition of the cost of a small wet storage facility reported by Dahlke et al. (Dahlke et al., 1994).

Spent Nuclear Fuel Storage Using Dry Casks:

Dry cask storage would include the use of concrete casks, both vertical and horizontal versions, metal casks, and multipurpose casks and would consist of the following components:

- A staging facility for cask receipt and unloading, and for loading foreign research reactor spent nuclear fuel into the dry storage casks. The staging facility would have a wet pool for unloading the casks and for short-term (1 to 3 years) storage of spent nuclear fuel with a heat load exceeding 40 Watts per element. This facility would include capabilities for drying the spent nuclear fuel/canister, inserting the spent nuclear fuel/canister with helium or nitrogen, and welding the storage canister closed.

- An inspection/characterization facility, for examining spent nuclear fuel integrity and canning leaking spent nuclear fuel as required. This may be incorporated into the staging facility (as an inspection cell) or be immediately adjacent to it. Although it is not expected that the physical condition of the foreign research reactor spent nuclear fuel elements would require extensive canning, the capability of canning the entire foreign research reactor spent nuclear fuel inventory would be provided by design.

- A dry storage cask (usually concrete). This would provide for the shielding and the structural stability of the spent nuclear fuel storage. The Multi-purpose Canister undergoing development could also be used (see Appendix F, Section F.1).

- A transfer mechanism, such as a dedicated truck/trailer combination with a ram for horizontal modules or a crane for vertical modules.

- A separate spent nuclear fuel canister may or may not be used. If used, it would typically be approximately 4.6 m (15 ft) long and 1.7 m (5.5 ft) in diameter, and would weigh approximately 33 metric tons (36 tons).

The dry cask approach would require the staging facility to receive and inspect the spent nuclear fuel shipment. The transportation cask would be unloaded in a small wet pool within the facility. Subsequently, spent nuclear fuel would be loaded into the dry cask (or spent nuclear fuel canister for the horizontal cask), and the cask would be placed on a concrete slab located outdoors. The horizontal approach would use a dry spent nuclear fuel transfer canister for containing the spent nuclear fuel. This would be placed within a shielded transfer cask and moved to the outside modular storage facility. A hydraulic ram would insert the transfer canister inside the horizontal storage module, followed by sealing with a shield plug. Thus, dry cask storage would always rely on the use of another facility.

Dry storage casks would be designed to withstand normal loads and design basis accident effects, such as earthquakes, tornadoes, and floods. Concrete would provide radiation shielding for gamma rays and neutrons. Natural air circulation would dissipate the heat; air would enter through inlet vents near the bottom of the cask, pass around the spent nuclear fuel canister, and exit near the top. Screens and grills would keep birds and animals out of the cooling duct area.

Some of the potential management sites have facilities which could be used for cask receipt and unloading and spent nuclear fuel inspection and transfer to storage. Utilization of these facilities would be considered.

The application of dry cask storage technology to foreign research reactor spent nuclear fuel would depend upon the heat load. Horizontal casks are anticipated to be slightly more restrictive than the vertical casks with respect to the heat load and are thus the focus of discussion. The standard design for a horizontal fuel canister would provide for 24 or 52 sleeves (i.e., pressurized water reactor or boiling water reactor spent nuclear fuel, respectively), each about 4.6 m (15 ft) long. As with the vault approach, it would be conservatively assumed that each sleeve contains 5 foreign research reactor spent nuclear fuel elements (i.e., in layers) within a basket or can arrangement for maintaining spacing and retrievability. Also, as with the vault approach, the number of dry storage casks would depend upon the decay heat of the spent nuclear fuel and a cladding temperature limit [nominally, 175°C (347°F) for aluminum-cladding with an air inlet temperature of 49°C (120.2°F)]. The 24-sleeve design would allow for a maximum of 120 elements for foreign research reactor spent nuclear fuel with 40 Watts to 80 Watts per element of decay heat, while the 52-sleeve design would provide for a minimum of 260 elements per dry storage cask with 10 Watts to 40 Watts per element. Thus, based on the total number of elements for which the facilities are sized, the number of casks required would be:

- ninety-four casks, predicated upon a 3-year cooldown period (i.e., less than 40 Watts per element). Note that this value is conservative and corresponds to a maximum of around 40 percent of the NRC-licensed heat loads per cask. Again, most foreign research reactor spent nuclear fuel is expected to have decay heats between 10 Watts and 40 Watts per element. Initially, foreign research reactor spent nuclear fuel with higher heat loads could be unsuitable for the dry storage cask pending detailed heat transfer analysis and a final determination of limiting fuel storage temperature for aluminum-based and TRIGA-type spent nuclear fuel. However, the relatively high decay heat spent nuclear fuel represents such a small percentage of the currently identified foreign research reactor spent nuclear fuel that its impact would be small, such that after 3 years of wet storage, it would all be below a heat output of 40 Watts per element.

Figure 2-9 displays the general layout for the dry cask storage facility predicated upon a horizontal cask design. Table 2-9 summarizes dry cask storage parameters.

Dry storage cask technology would require a separate staging facility for foreign research reactor spent nuclear fuel unloading, canning, and storage cask loading, and transportation cask maintenance. This facility would have the following operational areas:

- Transportation Cask Handling: this incorporates transportation cask maintenance, truck/railcar unloading, decontamination/washdown, radioactive material control, and cask sampling/flushing/degassing.

- A Small Wet Storage Pool: for fuel transfer and short-term storage.

- Spent Nuclear Fuel Unit Handling: fuel removal, decontamination, fuel drying, fuel canning, inserting with helium, and thermal measurements.

- Spent Nuclear Fuel Unit Transfer: this constitutes placement of the spent nuclear fuel into the cask or canister, followed by sealing.

- Radwaste Treatment: this includes collection, treatment, and preparation for disposal of contaminated effluents, and radioactive waste treatment and solidification.

- Heating, Ventilating, and Air Conditioning: this represents the component of the facility that helps ensure that contamination of workers and the environment is avoided.

The inspection/characterization facility would include a shielded dry hot cell for spent nuclear fuel analysis and examination, and canning of leaking spent nuclear fuel. All equipment and instrumentation within the cells would be remotely operated. The facility would be maintained under negative pressure with exhaust through high-efficiency particulate air filters to mitigate the environmental effects of any radionuclide releases. This facility is normally immediately adjacent to, or within, the staging facility.

Dry cask storage is unique among the three storage technologies because of its ability to be operationally integrated with existing facilities, which allows for faster implementation as compared to the other two storage technologies. Several management sites have facilities with spent nuclear fuel handling capabilities similar to the requirements of the staging facility. Potential examples include the Receiving

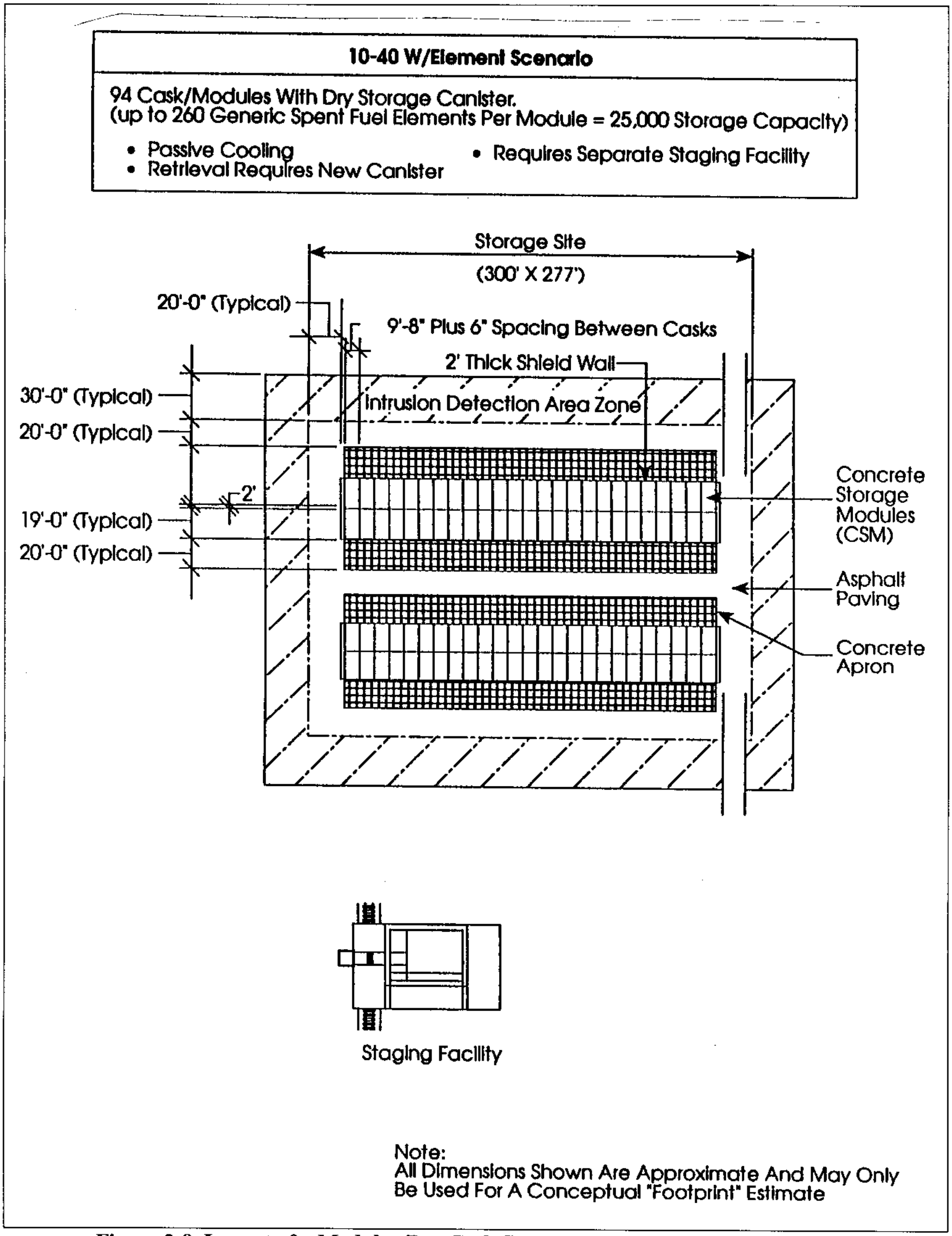


Figure 2-9 Layout of a Modular Dry Cask Storage Facility for Foreign Research Reactor Spent Nuclear Fuel (10 Watt to 40 Watt Element Basis)

Table 2-9 Summary of Dry Cask Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel[a]

Construction Phase:	
Disturbed Land Area	3 ha (7.7 acres)
Facility: size (area) concrete steel	 2,200 m^2 (24,000 ft^2) 17,500 m^3 (22,900 yd^3) 4,500 metric tons (5,000 tons)
Soil Moved	11,000 m^3 (14,400 yd^3)
Equipment Fuel	810,000 l (214,000 gal)
Construction Debris/Waste	1,800 m^3 (2,400 yd^3)
Work Force	50/yr for staging facility 50 per 24 cask array, 1 array per year
Duration (years)	5.5 for staging facility 4 years for construction, 1.5 years for design
Capital Cost	\$366 million[b]
Operation Phase:	
Electricity	800 - 1,000 MW-hr/yr (staging facility)
Water	2.1 million l/yr (550,000 gal/year) during receipt 0.9 million l/yr (238,000 gal/yr) thereafter
Wastestreams Solid Low-Level Waste Waste Water	 16 m^3/yr (565 ft^3/yr) during receipt 1 m^3/yr (35 ft^3/yr) thereafter 1.58 million l/yr (412,000 gal/yr) during receipt 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30 during receipt 8 thereafter
Annual Operating Cost	\$17.3 million during handling, \$0.3 million during storage[b]

[a] *Staging facility parameters are based upon the regionalized, small wet pool (Dahlke et al., 1994)*

[b] *Cost estimates are in 1993 dollars (EG&G, 1993)*

Basin for Offsite Fuels (RBOF) at the Savannah River Site and the CPP-666 storage pool area at the Idaho National Engineering Laboratory. For dry cask storage, the spent nuclear fuel would be shipped to the existing facility and unloaded from the transportation cask. The spent nuclear fuel would be inspected, canned if identified as a leaking element, and placed inside the storage canister. Spent nuclear fuel elements with heat loads exceeding 40 Watts per element would be stored in the existing facility to allow cooldown prior to cask storage. After filling, the canister would be sealed and placed inside the storage cask. The only new construction required would be the concrete storage pad (for vertical casks) or the concrete storage modules (for horizontal casks).

The cost to construct a dry cask storage facility with a staging area sufficient to unload, characterize, can, temporarily store in a small pool, and transfer the spent nuclear fuel to the cask storage area is estimated to be \$366 million. The annual operating cost for this facility is estimated to be \$17.3 million during the period of handling and transfers of the spent nuclear fuel and \$0.3 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993) with the addition of the cost of a small wet storage facility reported by Dahlke et al. (Dahlke et al., 1994).